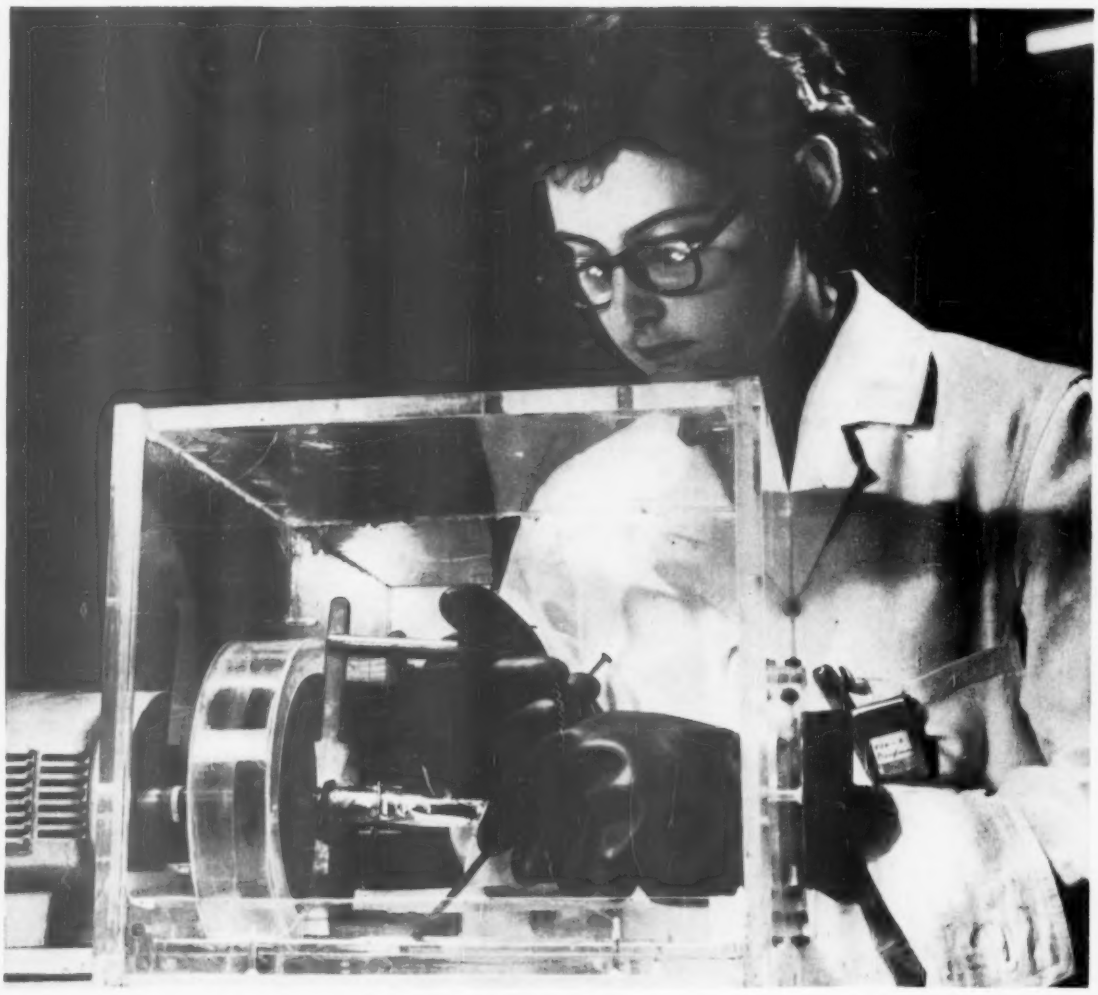
NUCLEONICS

January 1953



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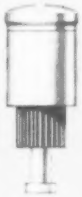

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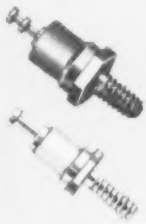

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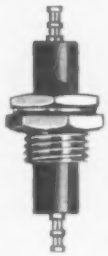

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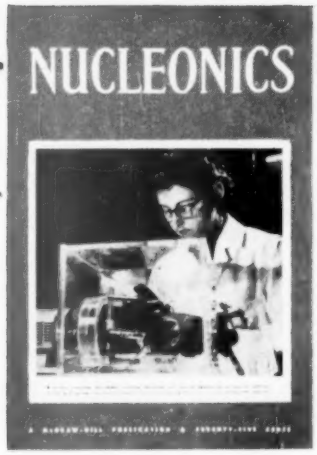

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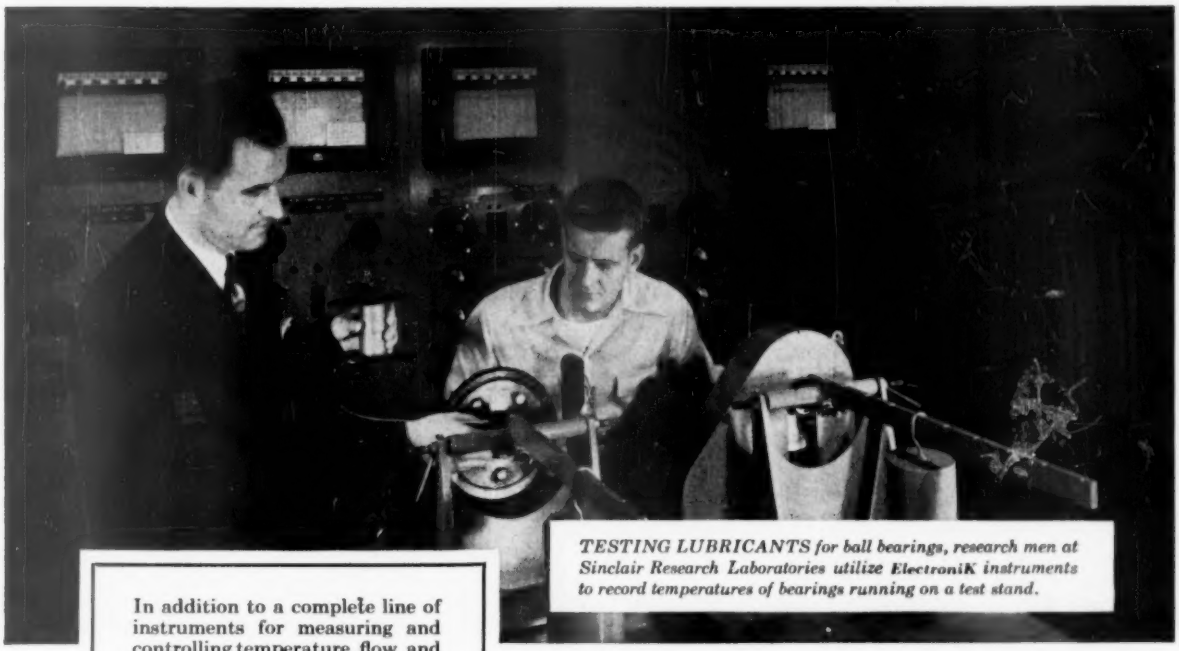

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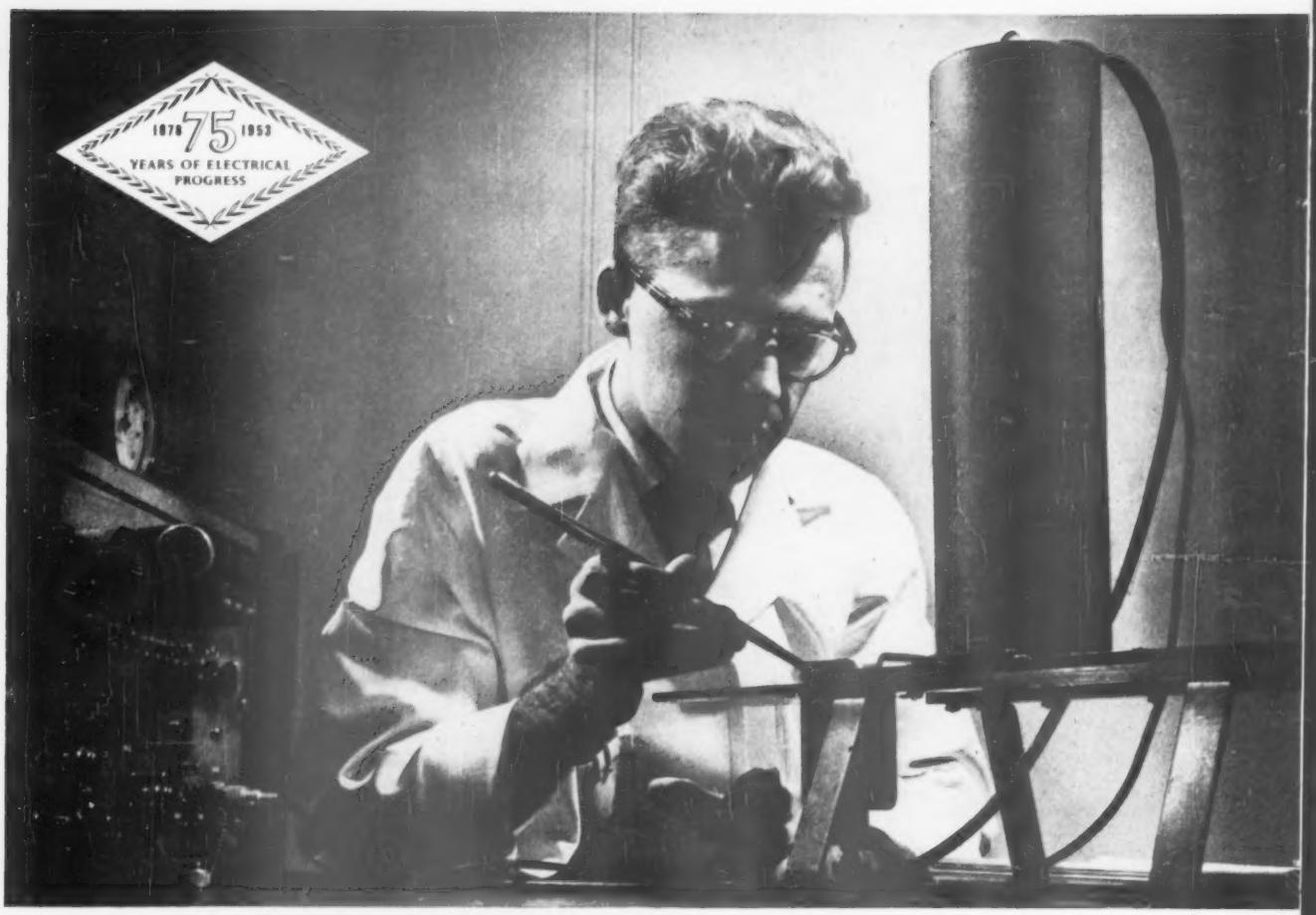

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GENERAL ELECTRIC

NUCLEONICS JANUARY, 1953

Jerome D. Luntz, Editor

A Need to Know

IN THE ATOMIC ENERGY FIELD, an individual is only cleared for access to classified information if he has a "need to know." He must be doing work for the Atomic Energy Commission or one of its contractors and must prove the need for information in some particular area of knowledge.

Over the past ten years, despite the fact that billions of dollars of public funds have been expended, the U. S. citizen has been kept pretty much in the dark on the important details of the program. The first major pronouncement came with the issuance of the Smyth report in 1945. Since that time quite a bit has been said and written about atomic energy but very little on such things as the production and uses of fissionable material—the heart of atomic energy developments.

In all these years, every major decision—including such things as the Savannah River and Ohio River multibillion dollar plants—has been made without the benefit of public commentary or examination. True, the Joint Congressional Committee on Atomic Energy has represented the public as a "watchdog," but the fact remains that the handful of members of the committee are also part of the "inner circle."

THE LAST PUBLIC DISCUSSION of any import was on the Atomic Energy Act of 1946. At that time, in several months of hearings, every public, industrial or governmental figure who had anything to say on the subject said his piece. Out of those hearings came an act which, although it is not perfect, has served the needs that existed at that time. We can be sure that if the various bills presented were debated behind closed doors Congress could not possibly have come up with a product representative of the desires of the public—an act which, until recently, has had very little serious criticism thrown at it.

Now we are at the point where we are on the eve of possible major legislative action in the atomic energy field. We are at the crossroads. Industry wants to take atomic energy out of the laboratory and into the production shop. But the Atomic Energy Act forbids this.

Mounting pressure appears to be forcing the issue. Congress and the AEC must decide whether or how to modify the act. Hearings of one kind or another are certain to be held this spring. The question is, "What kind of hearings?" Specifically, open or closed?

Unhesitatingly, we say "open." If never before, the public now has a "need to know." The rate at which nuclear power development will proceed in this country will be decided in the next months. The answers to such questions as these will be sought: What rights should industry have? Who should carry the financial burden in power development? What should the relationship be between the civilian and military programs? Etc.

For intelligent decisions to be made, it is absolutely necessary that the general and technical public know exactly where they stand in the nuclear power field. The recommendations of those closest to the subject must be made known. What do the commissioners of the AEC think? What do leading industrialists think? And what does the complete outsider think?

SECRECY need be no barrier to such open hearings. In a report issued last month, the Joint Committee on Atomic Energy briefly discussed twelve important questions on all aspects of "Atomic Power and Private Enterprise." Much more can be said on each one without compromising secrecy.

If, however, it is felt that more information is necessary before adequate hearings can be held, then the formula suggested on page 6 by J. G. Beckerley, AEC director of classification, could provide a basis for releasing data.

In any event, there appear to be no good reasons why the public's "need to know" cannot be satisfied.

Declassification Problems in Power Reactor Information

Major declassification of reactor data is necessary to foster growth of a reactor industry. Here AEC's director of classification evaluates end use of data in terms of "help to U. S. versus help to inimical interests"

By J. G. BECKERLEY
Director of Classification
U. S. Atomic Energy Commission
Washington, D. C.

FOR THE FIRST TIME since the Atomic Energy Act of 1946 was debated, the country finds itself in the midst of a vigorous discussion of how the act should be amended to meet present-day needs in atomic energy development. This discussion has been brought on by the active interest of industry in the possibilities of nuclear power.

It is apparent that all aspects of the future of power reactors will be the subject of public, Congressional and Atomic Energy Commission discussion during the coming months. One important question is that of secrecy. Two events last month point this up.

On December 15, Gordon Dean, chairman of the AEC, suggested the possibility that "major declassification actions" may be needed before any broad segment of private enterprise can effectively participate in the development of nuclear power.

Two days before that, the Joint Committee on Atomic Energy released a 415-page report on "Atomic Power and Private Enterprise." In citing the opinion of "most persons interviewed," the report stated that "the reactor art would advance faster and at less expense if the AEC could see its way clear to declassifying all pertinent data, including precise construction and production cost figures as well as information relating to fuel preparation and processing."

This spells out what undoubtedly would be necessary for a "major declassification" to be carried out. For the information to be used effectively in this country, it is necessary that the release of it not be made on a piecemeal basis. Substantial areas of information must be declassified to an extent that will be dependent upon the intended end use of the information.

The declassification of power reactor information will present a number of difficult problems. It is essential that the technical community, e.g., the readers of NUCLEONICS, understand these problems and give them proper attention, since public officials will seek and need technical guidance.

Classifying Reactor Information

The basis for keeping information on power reactors secret stems from the definition of "restricted data" in section 10 of the Atomic Energy Act:

> "SEC 10.(b) (1) The term 'restricted data' as used in this section means all data concerning the manufacture or utilization of atomic weapons, the production of fissionable material, or the use of fissionable material in the production of power, but shall not include any data which the Commission from time to time determines may be published without adversely affecting the common defense and security."

Two phrases are pertinent, namely, "production of fissionable material" and "use of fissionable material in the production of power." The first is applicable to plutonium or uranium-233 producing reactors while the second concerns power reactors. Information on power-breeders or power-plus-plutonium reactors are caught under both prohibitions and, thus, are doubly difficult to evaluate in declassification review. Information on power-only reactors would be relatively easier to justify for declassification.

The various reasons why Congress enacted a law specifically prohibiting the publication of information in the reactor field need not be reviewed here. Suffice it to note that, at the present moment, most technical power information is kept as "restricted data" because the Commission believes its release may be used to the detriment of the national security.

Reactor Information Categories

In the classification of information as "restricted data" the specific uses which are of concern to the AEC are those listed in the table (in approximate order of importance).

The precise use of classified technical information by a threatening nation is not deducible in advance. Just as knowledge on the part of company A of competing company B's process does not mean that company A will in any way alter its process, so it is uncertain whether a menacing "competitor" will alter in any way his reactor and fission-

able materials program, even knowing substantial data on ours.

The extent of his use of our information depends on many intangible factors such as the degree of his confidence in his own approach as well as such tangibles as the availability of materials and skills necessary to undertake U. S.-imitated projects. This uncertainty increases greatly the difficulty of declassification decisions. In fact, the usual conservative approach is to inquire into how much a potential enemy can *conceivably* use the information rather than into how much he will *actually* use it.

It should be noted that the list of possible "inimical interest" uses does not include the possibility of advancing his nuclear power production program, that is, his development of stationary power-producing units. Recent expressions of potential military interest in reactor "power packages" for remote or inaccessible military installations might indicate that we should include power-only units (nonmobile) among our sensitive developments. However, for purposes of the present discussion, we shall consider such units well down the list. Clearly we should be more reluctant to help a potential enemy in his fissionable-material and mobile-reactor projects than in any effort on his part to develop power sources.

Each type of reactor presents a different problem in any declassification review. Information on the design, construction and operation of a plutonium-only reactor may be useful to a competitor in his design, construction and operation of plutonium production reactors. It will be of secondary interest to him in his mobile reactor program. In addition, the manner of operating our plutonium production reactors will give him data on the specifications of the plutonium required for our weapons program. Power-only reactor information may be of use to a competitor in his mobile-reactor projects, although it is clear that he will need to develop minimum-weight shielding.

Some fast reactors are sufficiently akin to fission weapons that the release of technical data on them might reveal information of direct weapons use. Release of economic data on our plutonium production reactors, when coupled with publicly available budget and cost data, tends to compromise our rate of production of weapon-grade plutonium.

Even when specific numerical technical data is withheld from publication, there are some rather knotty declassification questions. Two of the most troublesome for plutonium-and-power units are (1) What can be released on plutonium costs? (2) What can be released on the desired specifications of plutonium?

The release of information on the current Commission cost of plutonium per se would be relatively harmless. However, it is likewise true that unless some description of the method of computing the cost were simultaneously released the usefulness of the figure would be small. It is essential in any public discussion of the pros and cons of various power-plutonium reactor proposals that adequate cost data be available. Certainly the estimated cost, in dollars per gram of produced plutonium, must be compared with what the Commission pays or is willing to pay for its currently produced material. If this comparison is made and if any indication is given of the method of cost computation, it will be possible by careful scrutiny of the Commission's public accounting of expenditures, as well as the current and future budgets, to estimate the current and future plutonium production. Even without budget data, knowledge of manpower and electrical power requirements, as well as capital costs, for any production site provides a good index to production costs. From the unit cost, production quantities are then readily deduced.

In addition to the question of unit cost release, there is the question of how much may be said on the specifications of plutonium when produced for weapon use. Unless a reactor is run properly, e.g., with correct flux and exposure time, the plutonium produced will not meet the specification set for "weapon-grade" material. This specification has technical significance to the weapon program. It is important information and not readily deduced, so that there is a real reluctance to declassify it. The fact that the specification ties in directly to the operating schedule of a plutonium-producing pile makes it particularly troublesome to conceal.

It is important to note that in both of those instances what is involved is not highly technical information. The dollar cost of plutonium is just "$x per gram." The operating data related to the specification of weapon-grade plutonium may in some cases reduce to a simple formula such as "run the reactor for d days at p kilowatts power level." Filling in the unknowns in these equations generally, for a given plutonium production reactor, will compromise information on the current atomic weapons program.

It is also important to note that the problem of compromising weapons and production information is not involved in the power-only reactors. In the power-only reactors plutonium costs are not involved and the operating schedules are strictly dependent on power requirements and fuel lifetime. As long as plutonium-power reactors are producing plutonium to be fed into the weapons program there are going to be instances where release of information on the reactor is equivalent

Possible Uses of U. S. Reactor Data by "Inimical Interests"

Type of assistance to "inimical interests"	*Type of reactor data*	*Possible use of data by "inimical interests"*
Technical	Plutonium (or U^{233}) production reactors, breeders—with or without power	Production of fissionable material for atomic weapons
Technical	Plutonium-power or power-only reactors, power breeders—mobile or nonmobile	Design and construction of mobile (military) reactors
Technical	Certain fast reactor data	Design of atomic weapons
Intelligence	Economic data on plutonium production reactors (with or without power)	Estimating U. S. plutonium production
Intelligence	All power-producing reactors, mobile or nonmobile	Evaluating status and potential of U. S. military reactor program

to release of information on weapons. Unless it is agreed that the weapons information involved is not of extreme sensitivity and that withholding it would place undesirable limitations on the public discussion of the pros and cons of plutonium-power reactors, it will be necessary to be relatively circumspect in discussing certain aspects of plutonium-power units.

Possible End Uses

In the declassification of power reactor information it would be pointless to release information in a random way. Even declassification in order of the increasing sensitivity of the information would be inappropriate. Rather, it is essential that a consistent body of information be released with a definite end use in mind. Without such purposeful declassification it is entirely possible that the end result would be the release of a heterogeneous group of facts of no real use to anyone, except possibly to competitors in filling in the gaps in their knowledge of the U. S. reactor program.

One way to define the various possible end uses of the declassified information is to consider the various groups who may use the data. It is convenient to think in terms of three such groups:

1. The general public as well as certain categories of industrial management who are interested in having sufficient information of a qualitative and nontechnical nature to stimulate a broad interest in participating in the reactor program as well as to permit a general evaluation of results.

2. Industrial management who will want sufficient information to evaluate the desirability of participating in the reactor program, for example, by building or operating power reactors.

3. Industrial engineers, designers and technical management who will want sufficient information to prepare preliminary design and feasibility studies including economic evaluations of power and power-plutonium reactors.

To satisfy the needs of the first group it is only necessary to release general conclusions of power reactor studies and similar qualitative information on design concepts, etc.

The second group can be satisfied only by providing more information on the technical features of power reactor designs. This information can possibly be minimized by the adoption of a "black box" reactor concept, in which information on what the reactor can do is released but information on precisely how the reactor is constructed to achieve its performance is withheld.

With respect to the third group, experience shows that the needs of this group can only be satisfied by providing almost all reactor information, possibly excepting data on minimum-weight shield technology.

For the second group, declassification to provide data for realistic appraisals of the pros and cons of participating in the development, construction or operation of power reactors may be possible under a "black box" concept.

Adopting this approach would mean that information such as the following would be released for particular reactor designs: primary coolant inlet and outlet temperatures; pumping power required for primary coolant circuit; dimensions of reactor vessel; shield thickness; all ancillaries, such as control circuits, heat exchangers, etc. (Designing certain of these might require access to classified information.) It is possible to provide such information and still withhold the critical and difficult details of the design of the heart of the reactor.

In addition, to serve those interested in plutonium-power reactors, plutonium cost and specifications would have to be released.

The information which would be withheld would be such items as: structure of fuel elements; detailed geometry of reactor core; corrosion or distortion of fuel and means for preventing reactor failure; details of design calculations, etc.

It should be understood, however, that this "black box" concept will not permit critical appraisal of the validity of certain information, for example, assessing whether the estimates of reactor performance are entirely reliable. It will be impossible to make completely valid comparisons between alternate designs without knowing more about the "black box's" contents.

Conclusion

It is apparent that, if the power reactor program is to be participated in by any broad segment of U. S. industry, some "major declassification actions" will be required. Such actions will probably furnish information of use to our unfriendly competitor in his reactor and atomic weapons program. The extent of such aid will depend on two factors: (1) the kind of reactors about which information is to be publicly disclosed and (2) the audience for whom the declassification is intended.

If power-only reactor data are released, the "aid and comfort" to inimical interests will be somewhat less than if power-plus-plutonium reactor data are declassified. If the purpose of declassification is only to stimulate broad interest and to provide minimum quantitative information, then the possible harmful effects of declassification will be considerably less than if the purpose is to provide design engineers with sufficient information to prepare preliminary reactor designs.

In any declassification action it is always possible to imagine the aid which the information may give to a potential enemy. This must be balanced against the similarly incalculable but extremely real benefits which this nation will achieve if a greater fraction of U. S. industry is well-informed and thereby capable of significant contributions to power reactor development. This country has grown strong and has prospered through competitive effort. Such competitive achievements will be seriously limited if the current declassification policy is retained. This appears to be a fairly unanimous opinion.

The key question is, "How far do we go beyond the present position?" It is perhaps foolhardy to attempt a prediction in this matter. However, I believe it is safe to say it will be done in stages with sufficient periods between stages to permit evaluation of the effects. The rate of progression through these stages will in turn depend very strongly on the rate of solving some of the basic legal questions of importance to industry—the questions of patent rights and plant ownership in the nuclear power field.

Although, as the Joint Committee on Atomic Energy has reported, there are many who feel that broad declassification will materially aid the present AEC reactor program, it may be difficult to justify the declassification of considerable data on power reactors unless it is clear that industry is ready to invest its own money in this business.

Thus, it would seem that, in conjunction with any considerations of changing the Act, it would be desirable to make enough information available to permit a broad evaluation by industry of its interest in nuclear power.

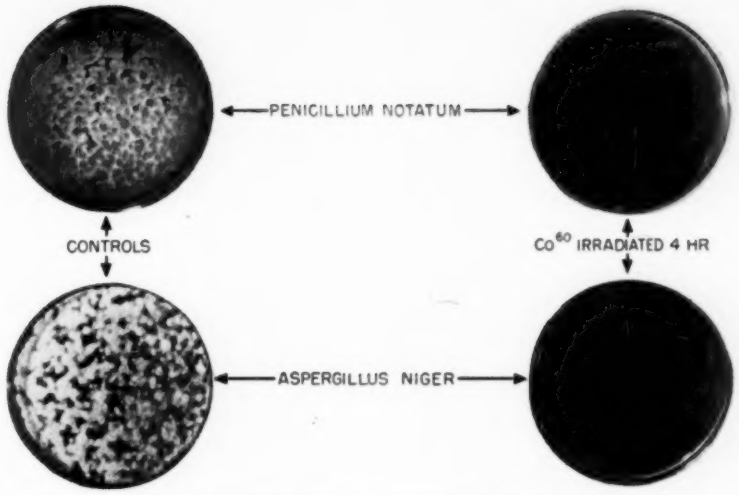

FIG. 1. Effect of cobalt-60 gamma radiation on molds

Effect of Cobalt-60 Gamma Radiation on Microorganisms

Cultures of bacteria, molds, yeast, and protozoa have been irradiated with a kilocurie source to demonstrate the ultimate usefulness of fission products for sterilization

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As part of investigations into possible industrial uses of waste fission products, the Fission Products Laboratory of the University of Michigan's Engineering Research Institute is studying the effect of radiations from its kilocurie cobalt-60 source as a sterilization medium for various organisms (*1–6*). Reported here are studies on non-spore-forming bacteria; spore-forming, aerobic bacteria; mold spores; yeast; and protozoa. Dose rate was calculated at 79,000 rep per hour in air. The four methods used to calibrate the source are discussed in the last section. All doses given are for measurement in air.

Non-Spore-Forming Bacteria

The organisms used consisted of 18-hour cultures of *Proteus vulgaris* and *Escherichia coli* grown on nutrient agar slants. The cells were washed twice with saline solution (0.85% sodium chloride); after centrifuging, the supernatant solutions were discarded and the cells finally suspended in saline solution containing 0.5% gelatin. Approximately 5 ml of the latter suspensions were distributed in 15 × 150 mm Pyrex culture tubes with screw caps and exposed to the gamma radiation.

Since previous calibrations of the temperature within the Co^{60} source indicated it to be comparable to the temperature of the room in which it was stored, the control culture suspensions which were not irradiated were maintained at room temperature in the laboratory during the period of irradiation of the test suspensions.

With but occasional exceptions, the dilutions of the treated and untreated organisms and the subsequent plating in agar were carried out within one hour after the cultures were removed from the cobalt source. In the pre-

* Information on the current status of this project appears on page 77.

TABLE 1—Microorganism Count as a Function of Radiation Dosage

Time of Irradiation	*Radiation dose millions of rep (in air)*	*Proteus vulgaris*	*Per cent reduction*	*Escherichia coli*	*Per cent reduction*
0 (control)	0	1,666,000,000	0	1,976,250,000	0
50 min	0.071	4,500,000	99.7	1,600,000	99.8
1 hr 45 min	0.149	3,680	99.99	500	99.99
2 hr 25 min	0.205	55	99.99	0	100.00
3 hr 40 min	0.289	0	100.00	0	100.00
5 hr	0.425	0	100.00	0	100.00
		Bacillus subtilis		*M-S spore-former*	
0 (control)	0	18,000,000	0	97,250,000	0
4 hr	0.34	960,000	94.6	8,750	99.99
8 hr	0.68	202,000	98.8	3,280	99.99
12 hr	1.02	20,500	98.8	550	99.99
16 hr	1.36	14,450	99.99	80	99.99
20 hr	1.70	75	99.99	10	99.99
24 hr	2.04	0	100.00	0	100.00
		Aspergillus niger		*Penicillium notatum*	
0 (control)	0	45,000,000	0	11,250,000	0
1 hr	0.085	125,000	99.66	15,000	99.86
2 hr	0.170	5,000	99.98	125	99.99
3 hr	0.255	25	99.99	0	100.00
4 hr	0.340	0	100.00	0	100.00

TABLE 2—Yeast Counts as a Function of Gamma and X-ray Dosages

Gamma radiation dose (cobalt-60). in millions rep (in air)	*Saccharomyces cerevisiae*				
	Viable cells per ml	*Per cent reduction*	*X-radiation in roentgens*	*Viable cells per ml*	*Per cent reduction*
0	2,750,000,000	0	0	3,250,000,000	0
0.085	200,500,000	92.72	86,000	55,250,000	98.30
0.070	26,250,000	99.05	132,000	10,000	99.99
0.255	32,000	99.99	218,000	9,000	99.99
0.340	25,000	99.99	344,000	5,500	99.99

liminary series of dilutions, 0.2 ml of the suspensions were added directly to the center of sterile petri dishes and 0.2 ml placed in 100 ml of sterile saline solution. Then 0.2 ml of the latter dilution was added to additional sterile petri dishes, and 0.2 ml of the same suspension was placed in a second bottle containing 100 ml saline solution. This same method of plating and dilution was carried out for an additional or third series. All suspensions were plated in duplicate, using approximately 15 ml of nutrient agar per plate. The plates were incubated at 37° C for 48 hours and the numbers of colonies developing in the medium were counted. The results of this experiment are presented in the first portion of Table 1.

The data show that a 99% reduction in viable organisms was obtained at the first sampling period after 50-min irradiation. Exposure for approximately 0.29-million rep resulted in sterile solutions of both bacterial suspensions.

Spore-Forming, Aerobic Bacteria

A four-day-old culture of *Bacillus subtilis* and a 22-hour culture of a spore-forming bacillus originally isolated from canned evaporated milk were used. Both were grown on nutrient agar slants. Suspensions of the organisms and their spores were prepared for irradiation as in the previous experiment. Microscopic examination of the suspensions before irradiation revealed an abundant number of spores.

The results of this experiment, presented in the second part of Table 1, indicate that the dose of gamma radiation required to destroy the sporulating organisms is much larger than that required to kill organisms of the non-spore-forming varieties. A radiation dose of 1.7- to 2.0-million rep is needed to destroy the spore-forming microorganisms completely. A 94.6% and 99.99% reduction in the numbers of viable organisms was obtained following 0.34-million rep radiation of *B. subtilis* and the spore-forming bacillus isolated from milk, respectively. Figure 3 shows the series of agar plates of the latter organism.

Figure 2 is a plot of the data from the first and second parts of Table 1. The use of $N + 1$ as the ordinate permits extrapolation to $N = 0$ or $N + 1 = 1$. This represents the mathematical requirements for complete sterility. The plot also indicates the effects of Co^{60} gamma radiation on the bacterial population of a sample of raw cow's milk; about 1.7-million rep are required to sterilize milk completely.

Mold Spores

Spore suspensions of two molds were prepared by washing the cells from 22-day-old cultures of *Penicillium notatum* and *Aspergillus niger*, grown on Sabouraud's maltose agar slants. The latter medium was also used for counting the residual numbers of viable spores following irradiation. In these instances, the mold counts were made after the plates were incubated at room temperature for five days. The results of this study are presented in the third part of Table 1. Unlike certain bacterial spores, the two species of molds have the same order of sensitivity to the lethal action of radiation as noted with non-spore-forming bacteria. Exposure period of 0.26- and 0.34-million rep resulted in zero counts for *A. niger* and *P. notatum*, respectively. Figure 1 shows the agar plates examined in this study.

Yeast

A yeast culture of *Saccharomyces cerevesiae* was grown on Sabouraud's maltose agar slants and the final plate

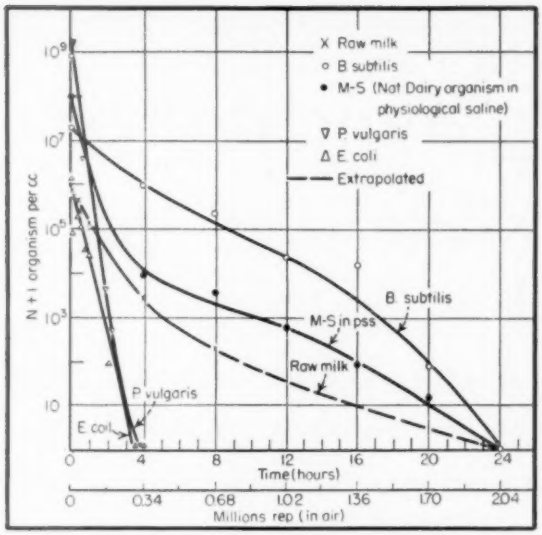

FIG. 2. Residual number of organisms plus one as a function of irradiation time by a 1,000-curie Co^{60} source

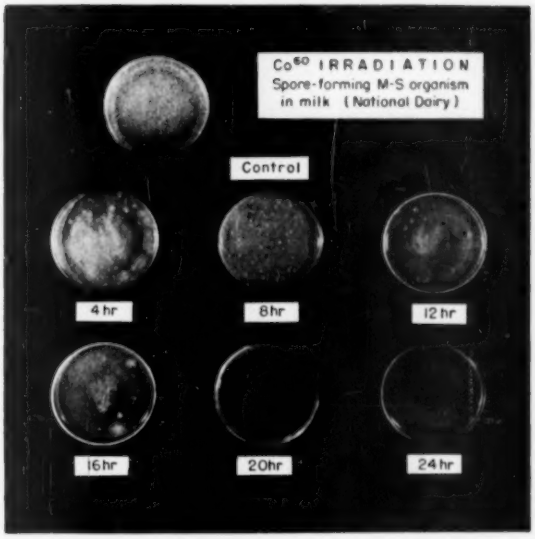

FIG. 3. Effect of gamma radiation on spore-forming (M-S) microorganisms in milk

counts also made in the same medium. Preparation of the suspension for radiation, plating, and counting the viable organisms were carried out as before.

The results of this experiment are presented in Table 2, which also compares the effects of gamma with that of X-radiation* on the same yeast suspension. The degree of lethal effects from gamma rays emitted from Co^{60} and from X-rays are comparable. While there appear to be some differences in the counts in the initial irradiated samples (0.085- and 0.170-million rep, respectively), the percentage reduction variations are within the limits of error encountered in a test of this nature.

Protozoa

One-milliliter amounts of a four-day culture of *Trichomonas foetus* (causative agent of "infectious abortion" in cattle) were suspended in a fluid portion of Schneider's medium (*7*) and exposed in 10 × 75 mm Pyrex tubes to Co^{60} gamma radiation. Motility and numbers of viable organisms were determined by examining the treated and untreated organism suspensions under the microscope. One-tenth milliliter of the samples was also inoculated into duplicate sets of tubes of Schneider's medium and checked for growth after three and four days' incubation at 37° C.

In all the suspensions which had been given 0.08- to 0.40-million rep doses, the organisms appeared to be actively motile. On the other hand, it is only in the higher doses (0.32–0.4-million rep) that there appears to be some indication of a more sluggish activity than that noted in the nonirradiated controls. Examination of the medium inoculated with the treated suspensions following incubation for four days showed growth in those tubes inoculated with the organism exposed to 0.09-, 0.17- and 0.26-million rep, but not in the cultures exposed to 0.32- and 0.40-million rep. The latter results indicate that while immediate death of the organisms is not observed to take place as the effect of irradiation, some damage to the cells occurs that prevents the organisms from multiplying—an aspect still being investigated.

Calibration of Gamma Source

The ferrous sulfate method of Weiss and others (*8*) was used in a preliminary calibration. The data gave a dose rate of 62,300 rep/hr (in water), using a value of 15.4 micromoles/liter/1,000 rep as recommended by Weiss. However, this calibration is being checked further.

A chloroform-water mixture was hydrolyzed by gamma radiation to produce hydrochloric acid and the latter used for calibration (*9, 10*). Hydrolysis by known dose rates (in air) was determined by *p*H measurements and compared with measurements made within the source. A value of 74,000 rep/hr (in air) was obtained.

Du Pont "Adlux" film and Victoreen r-meter chambers were compared with ionization chambers of the "Cutie Pie" type calibrated for gamma radiation. Readings were taken at a point of suitable dose rate outside the lead container and about one foot above the center of the source. The dose rate in this position was within the limited range of the "Cutie Pie" meters. Also at this position and dose rate, the "Cutie Pie" meters and r-meters yielded essentially identical results. The "Adlux" films were calibrated at this position. Only the films and r-meters could be used to measure the dose rate within the source because the flux was so large. Both the film measurement and the r-meter gave similar values with a mean of 79,200 ± 9,000 rep/hr (in air).

* 210-kvp Westinghouse X-ray machine calibrated with instantaneous roentgen ratemeter (Victoreen Model 510).

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Strong-Focusing Synchrotron —Design and Cost Data

A new focusing concept makes it possible to confine accelerator beams to very small apertures. This will considerably reduce magnet size and make multi-Bev machines feasible. Data for a 100-Bev synchrotron are presented

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THE STRONG-FOCUSING principle, by means of which it now seems possible to build particle accelerators as large as 100 Bev, was developed last summer at Brookhaven National Laboratory in the course of a design study for a larger, improved cosmotron.

The fact that the 3-Bev cosmotron did operate as planned, made it possible to speculate on further methods, smaller apertures, and larger orbit radii, with confidence that the particles would obey the equations of motion.

Out of these speculations by the author, together with Ernest D. Courant and Hartland S. Snyder, grew this new principle for the focusing of fast-moving charged particles.

THE STRONG-FOCUSING PRINCIPLE

The concept that led to the strong-focusing synchrotron came from a suggestion that the aperture could be reduced if a change in magnet positioning were made. In the cosmotron the C-magnets (shaded portion of Fig. 1) are all placed so that the closed part of the C is on the inside of the orbit. This back-leg C-magnet has the weakness that fringing fields increase and useful aperture is reduced as the iron approaches saturation.

It was suggested that the back-leg C-magnets be alternated with front-leg C-magnets in which the iron return circuits were on the outside of the orbit. This would make it possible to use shorter gaps and stronger fields to compensate for the weakening of the field in the outer portions of the aperture.

Such an alternating-C design would have other advantages, such as making sectors of the inside of the vacuum chamber available for experiments on positive mesons produced at a target in the chamber. However, to make the average guide field uniform across the width of the vacuum-chamber aperture, the magnetic-field gradient would necessarily be different in the successive sectors. In the language of accelerator designers, the n-value would be different, positive in one sector and negative in the next, at least in the outer portions of the aperture.

The n-value is the exponent that describes the radial decrease in magnetic field required for focusing in magnetic accelerators, and is defined as

$$B = B_0(r_0/r)^n \text{ or } n = -\frac{r_0}{B_0}\frac{dB}{dr} \quad (1)$$

where r_0 and B_0 are the radius and magnetic field at the center of the vacuum chamber.

For stability in the betatron or synchrotron, $0 < n < 1$, which means a field which at one limit is uniform ($n = 0$) and at the other limit decreases inversely with increasing radius. Most synchrotrons and betatrons are designed for $0.25 < n < 0.75$. For the synchrocyclotron $0.05 < n < 0.2$.

The n-value must be positive, which means a field which is concave inwards, to produce stability for vertical oscillations, or for divergent ions in the z-coordinate.

In fields with uniform n-values around the orbit, the n-value must be less than unity for radial stability; that is, below this limit the centripetal force can counteract the divergence in the radial coordinate. Within these limits, both radial and vertical divergences are restored, or focused, about the central equilibrium orbit.

A basic principle in magnetic fields, coming from the Maxwell equations, states that in a single section of such a conservative field, particles cannot be simultaneously converged in the two transverse coordinates unless there is an additional force such as the central force in circular motion noted above.

Oscillations about the equilibrium orbit are known as "betatron" or

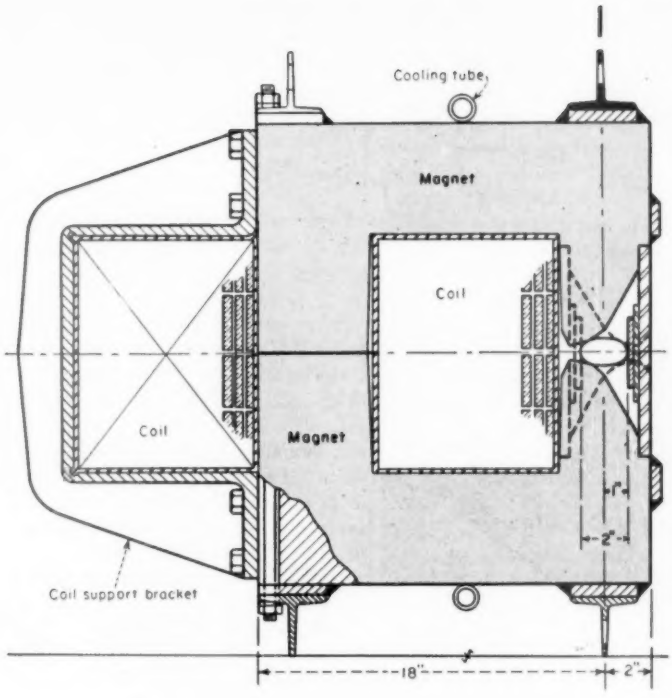
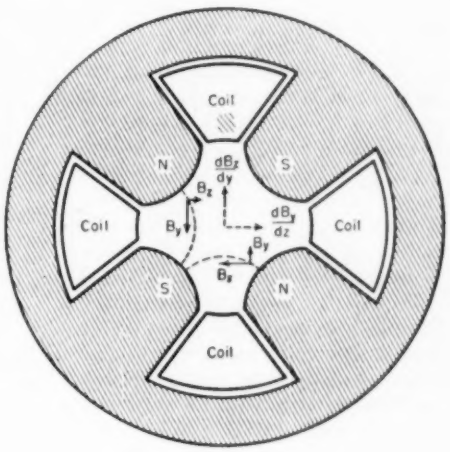


FIG. 1. (Left) Cross section of magnet designed to give large transverse field gradients over 2 × 1½-in. aperture. For focusing, alternate solid and dotted pole faces would be used

FIG. 2. (Below) Cross section of 4-pole magnet with hyperbolic pole faces to produce uniform and equal field gradients in the x- and y-directions



"free" oscillations. The frequencies of such oscillations are given by

$$f_s = n^{1/2} f_0; \quad f_r = (1-n)^{1/2} f_0 \qquad (2)$$

where f_0 is the orbital frequency of motion of the ions. For the simple case $n = 0.5$, both frequencies are equal to $0.707 f_0$, which means that the wavelength of the oscillations is longer than one complete turn. This relatively low free-oscillation frequency leads to large amplitudes and requires very well collimated beams from the source, a large cross section for the vacuum chamber, and a large magnet to supply the field over this aperture.

Now to return to the strong-focusing principle. The first question to be answered about alternating-C magnets having successively positive and negative n-values, is whether the stability of the free oscillations is affected. The answer found is that such stability is greatly improved. To understand this we must analyze the equations of the oscillatory motion (θ = azimuthal angle)

$$\frac{d^2z}{d\theta^2} + nz = 0; \frac{d^2r}{d\theta^2} + (1-n)r = 0 \qquad (3)$$

where there are N magnetic sectors around the orbit with n-values alternating between n_1 for odd sectors and n_2 for even sectors. Solutions to these equations can be obtained by recursion formulas, and are of the form

$$Z_k = Ae^{i2\pi\mu_s k}; \; R_k = Be^{i2\pi\mu_r k} \qquad (4)$$

where the coefficient μ_s is given by

$$\cos 2\pi\mu_s = \cos\frac{2\pi n_1^{1/2}}{N}\cos\frac{2\pi n_2^{1/2}}{N} - \frac{n_1+n_2}{2(n_1n_2)^{1/2}}\sin\frac{2\pi n_1^{1/2}}{N}\sin\frac{2\pi n_2^{1/2}}{N} \qquad (5)$$

The coefficient μ_r is given by the same expression, with n_1 and n_2 replaced by $(1-n_1)$ and $(1-n_2)$, respectively. For the motion to be stable for radial and vertical oscillations

$$-1 < \cos 2\pi\mu_s < 1$$
$$-1 < \cos 2\pi\mu_r < 1 \qquad (6)$$

The stability limits can be plotted for any chosen value of N, the number of alternating magnetic sectors, as a function of n_1 and n_2. At $N = 10$, for example, the center of the region of stability is for $n_1 = +6$ and $n_2 = -5$. However, it is soon evident that the focusing is strengthened by using very large values of N and n, where the difference between $|n|$ and $|n-1|$ becomes negligible and the amplitudes of both radial and vertical oscillations become equal and very small.

The range of stable values is widest when N is large and when $n_2 = -n_1$, and the center of the region of stability ($\cos 2\pi\mu = 0$) occurs for

$$|n| = N^2/16 \text{ or } N = 4|n|^{1/2} \qquad (7)$$

The stability in both radial and vertical free oscillations obtained by using large values of n and N increases the frequency of these oscillations by a large factor. The frequency is

$$f_s = f_r = (\mu N/2) f_0 \qquad (8)$$

The corresponding wave length is reduced proportionately, and the particles perform many radial and vertical oscillations in a single turn.

In a practical design, the sectors must be separated by short gaps to allow space for magnet end-windings, radiofrequency accelerator units, and vacuum pumps. The equations derived above, and the stability limits, will be slightly changed, but not significantly, if the gaps are no greater than 25% of the sector length.

DATA FOR A 100-BEV ACCELERATOR

As a numerical example, the design parameters can be cited for a synchrotron designed for 100 Bev (33 times the top energy of the cosmotron). Orbit radius would be 800 ft in an average guide field of 14 kilogauss. The vacuum chamber would be 2 in. in radial aperture and about 1.5 in.

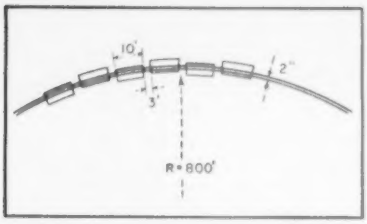


FIG. 3. Arrangement of alternate inside-C and outside-C magnets

vertically. The maximum field gradient attained in this aperture is 14,000 gauss/in. The n-value computed for this radius from Eq. 1 is 9,600.

From Eq. 7, $N = 392$; however, this must be modified somewhat when the effect of straight sections between sectors is included, and the value chosen is 480 sectors. Each sector would be 10.5 ft long, separated by 3-ft straight sections. The over-all average radius of the magnet structure, including straight sections, is 1,030 ft and the circumference of the polygon is about 6,480 ft.

The frequency of free oscillations in this field, from Eq. 8, is $60f_0$ (neglecting straight sections), so the amplitude is reduced to a few percent of that expected in a synchrotron with a constant value of $n \approx 0.5$. In physical dimensions, this means a double-amplitude width for such oscillations of about 0.5×0.5 in., which is still small compared to the aperture dimensions and leaves a safety factor of 3.

Magnet Design

The magnet to produce this field has strange looking poles, designed to obtain the largest possible uniform field gradient over the width of the aperture. It is based on the double-magnet shown in Fig. 2 which has four pole faces in the form of rectangular hyperbolas. The field is zero at the center and has a uniform and equal gradient over the entire volume between poles if the pole faces are magnetic equipotentials. The gradient is

$$dB_r/dz = dB_z/dr = \text{constant} = \frac{\text{field at "half-gap" location}}{\text{length of "half-gap"}} \quad (9)$$

If this magnet is split down the center-line axis and one side replaced by iron, the field pattern is unchanged. This can be called a 3-pole magnet with a north pole, a south pole, and a "neutral" pole the function of which is to preserve zero field at the iron surface. A practical modification of this shape to include suitably-shaped coils for excitation is shown in Fig. 1.

The aperture can be fitted with a $2 \times 1\frac{1}{2}$-in. oval vacuum chamber, formed of a nonmagnetic metal tube.

Consecutive magnet sectors (separated by straight sections) would be alternated with inside and outside return circuits, aligned on the vacuum chamber as shown in Fig. 3. The alternation of successive sectors can also be obtained by simply reversing the pole-face shape from the solid- to the dotted-line positions shown in Fig. 1. With alternating pole-tips, the return legs can be kept either outside or inside the orbit for sufficiently long runs to allow space for injection or for emergent particle beams.

Synchronous Oscillations

The accelerator described here operates on the principle of a proton synchrotron, in which the particles perform phase-stable synchronous oscillations in energy about an equilibrium value which increases with time, and follows the increasing magnetic field. The steady increase in equilibrium energy required can be computed from the magnetic-field cycle, and is found to average 770 kv per turn for the 100-Bev accelerator, if the rise time of the magnetic field is 1 sec.

These synchronous oscillations are identical with those of the cosmotron, except that the relation between particle energy and circumference of the equilibrium orbit is changed for particles with energies above or below the value at the central orbit. The equilibrium orbit for such a particle is no longer circular (with inserted straight runs), but consists of a periodic function of harmonic and hyperbolic sections superimposed on a circle of different equilibrium radius.

The orbits for high- and low-energy particles are illustrated in Fig. 4, in which the curvature of the central orbit is neglected, so that it appears as a straight line. The displaced dotted lines represent circular orbits of larger and smaller radii, along which the average value of magnetic field is the same if the field gradients in successive sectors are equal and opposite. However, the average field along the actual displaced orbit (solid line) is higher than that of the dotted line in both sectors for the upper curve (energy excess) and smaller in both sectors for the lower orbit.

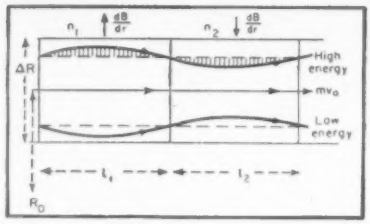


FIG. 4. Stable synchronous orbits in radial aperture

So particles of higher energy actually traverse fields averaging higher in intensity than the fields in the central orbit, and vice versa. This results in compaction of particle energy into a small radial spread about the central orbit. The change in circumference of orbit with particle momentum has been computed for these displaced orbits. The result can be expressed in terms of the fractional change in momentum associated with the fractional change in orbit radius. It can be shown that the 2-in. radial chamber width will accommodate a momentum spread, $\Delta p/p$, of $\pm 10\%$. Thus, large errors can be tolerated in the energy of the injected particles and in the frequency of the r-f accelerating units during early stages of acceleration.

Radiofrequency Acceleration

For the large orbit radius envisaged, the frequency of particle revolution in the orbit becomes small. For example, at an injection energy of 12 Mev, the orbital frequency is 26 kc, and it rises to 164 kc at the final energy. The required voltage per turn is too great to be applied in one step, as in the cosmotron, but should be distributed between many units located around the periphery. If the frequency of the r-f acceleration is a high harmonic of the orbital frequency, these many drive units can be phased alike and controlled from a central unit.

A large number of drive units, with a lower voltage on each, is also advantageous from the point of view of power requirements. It is a well-known principle of radiofrequency engineering that the total power required for n units, each supplying a voltage V/n, is $1/n$ times the power required for a single unit supplying a voltage V. These considerations have led to a design choice of 120 for the harmonic order to be used for the r-f acceleration.

The applied frequency would be 3.1 Mc at injection and 19.7 Mc at

maximum energy, a frequency ratio of 6.2:1. At this frequency there would be 120 bunches of particles uniformly distributed around the orbit. The number of drive units chosen is 240, half phased at zero degrees and the alternate units at 180 degrees.

One of the most important consequences of this increase in applied frequency by harmonic drive is the corresponding decrease in the radial amplitude of the oscillations associated with the synchronous energy oscillations. A calculation of the maximum amplitude of such synchronous oscillations, under these conditions, gives a radial extent of ± 0.25 in. This also is small relative to the radial extent of the chamber, and allows a safety factor of 3 for energy variation due to frequency error.

One possible method of producing the accelerating potential of about 6.4 kv per unit (twice the average requirement to allow phase oscillations) over the range from 3.1 to 19.7 Mc, is to use a ferrite-loaded resonant cavity driven by a broad-band power amplifier.

Ferrites have been used in a similar way in the cosmotron, and it is known that types exist which have adequately high magnetic permeabilities at frequencies up to 20 Mc. This figure determines the choice of maximum frequency and so the choice of the harmonic order discussed above. Noninductive, d-c bias windings on the ferrite core can be used to reduce the permeability by a factor of 40, which increases the resonant frequency of the ferrite-loaded cavity by $40^{1/2}$. So it is possible to tune the cavity over the frequency range without loss of electrical efficiency.

The copper cavity would be drum shaped, with an insulated section of the vacuum chamber threading it on the axis. The dimensions of the drum would be determined by the low-frequency limit when loaded with the unbiased ferrite; a reasonable dimension would be a diameter of 4 ft and a thickness of 2 ft, which would fit comfortably into the 3-ft straight sections. Model studies of ferrites are needed to determine the practicability of this method and the final dimensions.

It is estimated that such a loaded cavity would have a minimum Q, or resonant efficiency, of 10 over the frequency range. With this assumption, the power requirements can be estimated. Total power required for the 240 units is of the order of 1,000 kw, and could be obtained from a motor-flywheel-generator-rectifier using ignitrons similar to that used for producing the pulsed magnet power.

Vacuum Chamber

One of the most significant advantages of this design over the cosmotron is the simplicity of the vacuum chamber. A thin-wall stainless-steel tube of oval cross section will enclose the necessary aperture. Eddy currents induced in the wall would have such a small effect in altering the shape of the field as to be negligible. High-vacuum techniques, such as outgassing by heating under vacuum, are possible with such an all-metal chamber.

The insulated sections through the r-f cavities can be of glass or ceramic. The number of joints and the length of gasket-sealed joints is small. Pressures of less than 10^{-5} mm of Hg can be expected, with relatively small and inexpensive pumps.

Radiation Shielding

The building to house such a large-diameter accelerator must include radiation shielding as a fundamental concept of design. At the beam intensities expected (similar to those in the cosmotron), the minimum shielding in the horizontal plane of the beam would be about 20 ft of earth.

An obvious design concept is a trench-type building with the machine below ground level, and with a thick roof over the trench to attenuate scattered radiation. Railroad-type hoists on rails spanning the trench could be used for assembly.

A cluster of facilities would be grouped around one part of the orbit, including controls, injector (12-Mev linear accelerator), magnet, r-f power generators, maintenance shops, and target observation areas. Vacuum pumps would be distributed around the circle, possibly one small pump at each straight section. Magnet sectors, r-f power units, and pumps would be readily replaceable for maintenance; quick connections allowing a new unit to be installed with a minimum shut down of operations are essential.

Observation rooms to study the radiations from targets would also be below ground level, separated by an adequate earth fill for shielding, connected by pipes through which the beams emerge from the main trench, and located both inside and outside the orbit. New concepts in building design will be required to anticipate all the problems involved.

Cost Estimates

The preliminary estimating of costs is bound to be rough for a machine of this type, but many of the components can be compared with equivalent units in the cosmotron. Magnet cross section is much smaller, and iron costs are decreased by a large factor on a unit-length basis. The most expensive magnet component is the copper for the exciting coils, which are made large to reduce power requirements and avoid the necessity of water cooling of the coils. By choosing a rather large weight of copper, the total power can be reduced to about the same as that used for the cosmotron.

The radiofrequency units require a considerable tonnage of ferrite, and the power amplifier unit to drive each cavity will have a rating of about 10 kw. The large number of units increases installation cost but again decreases power requirements so that the primary d-c power supply is only a little larger than the magnet generator. Vacuum-pump units are small in capacity, and can be produced cheaply in quantity.

The large number of identical magnet sectors, radiofrequency units, and pumps suggests the use of production-line techniques for construction, probably by competent commercial firms. Design costs will be relatively low, since only one prototype of each component need be prepared. Assembly, following a flow-delivery system from the commercial suppliers, will also be a small fractional cost, although stretched over a long period of time.

The design problem as a whole seems easier than that of the cosmotron, and will not require as large a team of scientists and engineers.

The best guesses on costs over the 10–100-Bev range suggest a construction cost of approximately $300,000 per Bev, decreasing somewhat for the higher energies. The cost per volt is reduced to about $\frac{1}{10}$ of that for the cosmotron, for the higher energy range.

* * *

The original work on the strong-focusing principle was published in an article by Ernest D. Courant, M. Stanley Livingston, and Hartland S. Snyder, in Phys. Rev. **88**, *1190 (1952). Figures 2, 3, and 4 are reproduced through the courtesy of the editors of the* Physical Review.

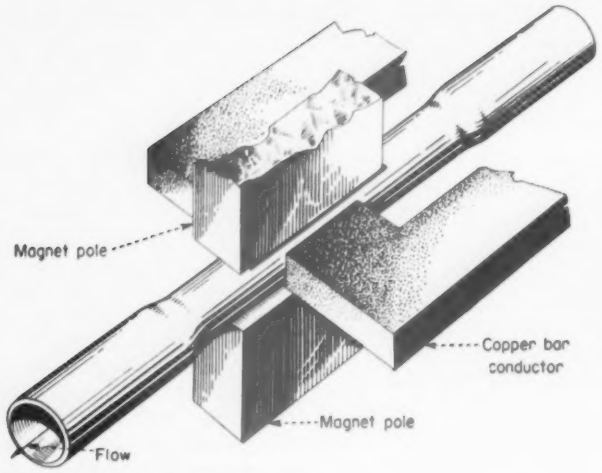


FIG. 1. A d-c electromagnetic pump

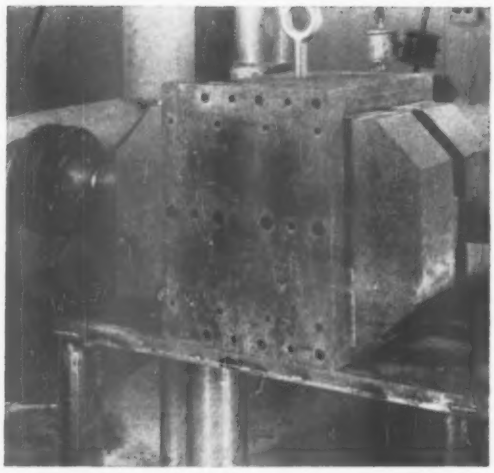


FIG. 2. Electromagnetic pump for primary coolant circuit of Experimental Breeder Reactor

Direct-Current Electromagnetic Pumps

Electromagnetic pumps have many advantages for pumping liquid metals. Principles of operation and the design of pumps are described, including the one used in the primary coolant circuit of the Experimental Breeder Reactor

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THE PRESENT INTEREST in liquid metals as reactor coolants has served to spur the development of pumping equipment capable of handling such materials at high temperatures. The electromagnetic pump, with its absence of bearings and seals, is a very attractive device in the face of the difficulties encountered in attempting to confine sodium or sodium-potassium alloy by any form of shaft seal together with the problem of designing bearings to operate in these liquids.

Since the pioneer work of Northrup in 1907 (*1*), many investigators have designed a wide variety of apparatus for pumping liquid metals by electromagnetic means. Work on electromagnetic pumps was begun at Argonne in 1946 (*2*) when several small, a-c induction (centrifugal type) pumps were tested. Subsequently, a linear induction pump (*3*) with a capacity of 100 gal/min was constructed.

This article will be confined to a discussion of the d-c electromagnetic pumps (*4*) which have been under investigation at Argonne since 1947. A discussion of early work is given in a series of reports of the Reactor Engineering Division (*5*).

Principle of Operation

The principle of operation of the direct-current pump is illustrated in

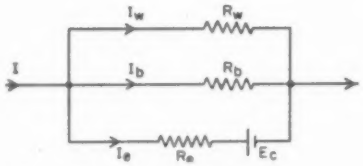

FIG. 3. Equivalent circuit of pump

Advantages of Pump

1. Completely sealed, all-metal system
2. No moving parts, no bearings to wear or require attention
3. Can pump liquid metals at high temperatures
4. No vibration
5. No radiation damage since practically no electrical insulation required

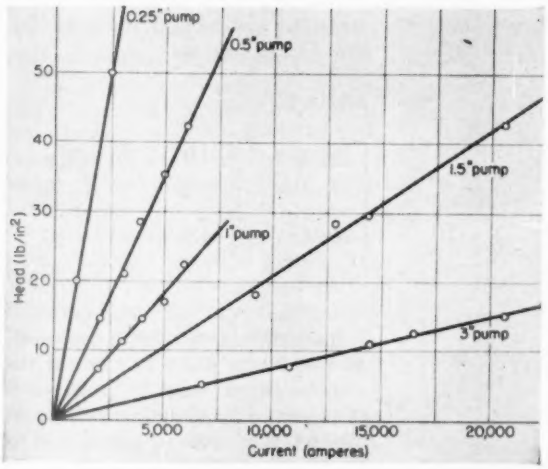

FIG. 4. Static pressure as a function of current for constant magnetic field

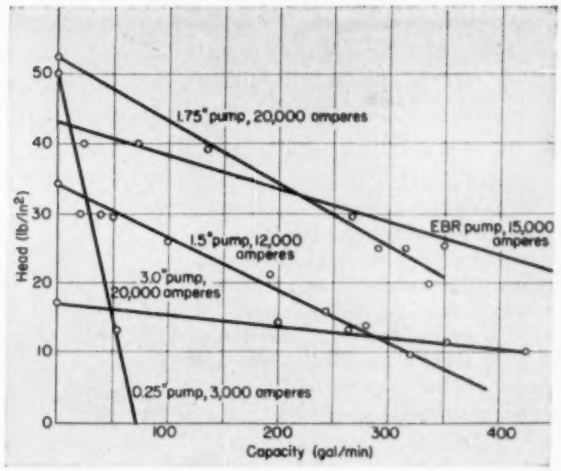

FIG. 5. Head-capacity characteristics of several pumps with constant current

Fig. 1. The pump used in the Experimental Breeder Reactor is shown in Fig. 2. The liquid metal flows through a thin-walled rectangular tube of stainless steel or other metal of high electrical resistivity. Copper bars are brazed to opposite sides of the tube, and the assembly is placed between the poles of an electromagnet. Current entering through the wall traverses the liquid in the tube and develops in it a longitudinal thrust.

The magnitude of the force on the liquid is

$$F = BI_e r/10 \text{ dynes} \qquad (1)$$

where B is the magnetic flux density in gauss in the liquid between the magnet poles, I_e is the current in amperes traversing the liquid which lies in the magnetic field, and r is the width in centimeters of the rectangular tube parallel to the direction of current flow.

If s is the width of the tube in the magnetic field direction, then rs is the cross-sectional area of the tube, and the pressure developed will be

$$P = \frac{BI_e r}{10rs} = \frac{BI_e}{10s} \text{ dynes/cm}^2 \qquad (2)$$

The total current traversing the tube will be the sum of the currents in the liquid and in the tube wall. The current in the liquid will consist of two portions: that which passes through the region of strong magnetic field and is therefore effective in developing a thrust in the liquid, and that which by-passes the region of strong field and contributes little or nothing to the pumping action.

Figure 3 shows the equivalent circuit of the system. The total current, I, is

$$I = I_w + I_b + I_e \qquad (3)$$

where I_w is the current which crosses the tube through the wall, I_e is the current traversing the liquid which lies in a strong field region, and I_b is the current traversing the liquid which passes through regions of weak field at entrance and exit portions of the tube.

Since voltage drops along parallel paths are equal

$$I_w R_w = I_e R_e + E_c \qquad (4)$$

$$I_b R_b = I_e R_e + E_c \qquad (5)$$

where E_c is the counter-emf developed in the liquid as it moves through the magnetic field, R_w is the resistance of the by-pass path through the tube wall, R_e is the resistance of the effective current path through the liquid, and R_b is the resistance of the by-pass path in the liquid.

Combination of Eqs. 3, 4, and 5 gives

$$I = I_e R_e \left(\frac{R_w + R_b}{R_w R_b}\right) + I_e + E_c \left(\frac{R_w + R_b}{R_w R_b}\right) \qquad (6)$$

Since $E_c = Brv \times 10^{-8}$ volts, where v is the velocity of the liquid in centimeters per second, and $v = Q/rs$, where Q is the flow through the tube in cubic centimeters per second, then

$$E_c = \frac{Brv}{10^8 rs} = \frac{BQ}{10^8 s} \text{ volts} \qquad (7)$$

Now, from Eq. 2, $I_e = 10Ps/B$, and substituting this and Eq. 7 into Eq. 6, and solving for Q we get

$$Q = \frac{10^8 s}{B}\left[I\left(\frac{R_w R_b}{R_w + R_b}\right) - \frac{10Ps}{B} \times \left(R_e + \frac{R_w R_b}{R_w + R_b}\right)\right] \text{ cm}^3/\text{sec} \qquad (8)$$

This relation shows how the pump capacity will vary with current, magnetic field intensity, and geometry of the tube. The values for R_w and R_e may be calculated from the dimensions and resistivities of the tube and liquid. The resistance R_b is best evaluated on the basis of experimental tests, since it is a complicated function of tube and magnet geometry, field distribution, and liquid velocity profile.

Equation 8 also indicates that there is a minimum value of current for which the flow is zero at any given pressure. Further, since

$$\frac{\partial Q}{\partial I} = \frac{10^8 s}{B}\left(\frac{R_w R_b}{R_w + R_b}\right)$$

the flow will increase linearly with current.

The expression for the static pressure developed is obtained from Eq. 8 by setting $Q = 0$, and is

$$P_{\text{static}} = \frac{BI}{10s} \times \left[\frac{R_w R_b}{R_w R_b + R_e(R_w + R_b)}\right] \text{ dynes/cm}^2 \qquad (9)$$

which is equivalent to Eq. 2 with the pressure now given as a function of the total current fed into the tube. If B

is expressed in kilogauss, I in kiloamperes, and s in inches, and if R_e is small compared to R_w and R_b, then Eq. 9 takes the form

$$P = \frac{0.57BI}{s} \text{ lb/in}^2 \qquad (10)$$

Figure 4 shows the variation of static pressure and current obtained experimentally with pumps of various dimensions. The magnetic field intensity was held constant in each case. The pumps are designated by the dimension of the pump tube, s, which is parallel to the magnetic field.

Rearranging Eq. 8 we have

$$P = \frac{BI}{10s}\left[\frac{R_w R_b}{R_w R_b + R_e(R_w + R_b)}\right] - Q\left[\frac{B^2}{10^9 s^2\left(R_e + \dfrac{R_w R_b}{R_w + R_b}\right)}\right] \qquad (11)$$

which indicates that the pressure-capacity characteristic (neglecting hydraulic losses) of the pump should be linear, with the pressure decreasing at the rate of

$$\left[\frac{-B^2}{10^9 s^2\left(R_e + \dfrac{R_w R_b}{R_w + R_b}\right)}\right] \frac{\text{dynes}}{\text{cm}^2} \Big/ \frac{\text{cm}^3}{\text{sec}}$$

as the flow increases.

Figure 5 illustrates this relation. These curves were obtained by allowing the flow to increase while the current and field were kept constant.

Again referring to Eq. 8, it appears that there should be an optimum value of magnetic field strength for which the flow is a maximum at a given pressure. Thus, setting

$$\frac{\partial Q}{\partial B} = \frac{-10^8 Is}{B^2}\left(\frac{R_w R_b}{R_w + R_b}\right) + \frac{2 \times 10^9 Ps^2}{B^3}\left(R_e + \frac{R_w R_b}{R_w + R_b}\right) = 0$$

one obtains

$$B(\text{optimum}) = \frac{20Ps}{I} \times \left[1 + \frac{R_e}{R_b} + \frac{R_e}{R_w}\right] \text{gauss} \qquad (12)$$

Figure 6 illustrates this effect.

The potential difference across the pump tube (i.e., between points at which the copper bars attach to the tube wall) will be

$$V_t = E_c + I_e R_e = \frac{BQ}{10^8 s} + \frac{10PsR_e}{B} \text{ volts}$$

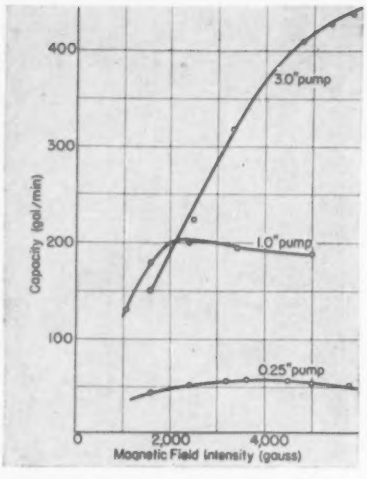

FIG. 6. Effect of magnetic field on capacity with constant current

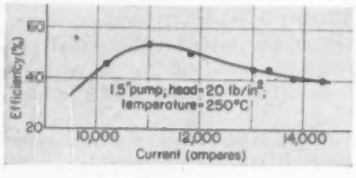

FIG. 7. Variation of efficiency with current

The electrical power input to the pump tube will therefore be

$$W_t = IV_t = I\left[\frac{BQ}{10^8 s} + \frac{10PsR_e}{B}\right] \text{ watts} \qquad (13)$$

The mechanical power transferred to the moving liquid is $10^{-7}\,PQ$ watts; so the efficiency of power conversion is

$$\text{Efficiency} = \frac{10^{-7}PQ}{I\left[\dfrac{BQ}{10^8 s} + \dfrac{10PsR_e}{B}\right]}$$

Substituting the value of Q from Eq. 8, this becomes

$$\text{Efficiency} = \frac{10Ps}{BI} \times \left[\frac{IR_w R_b - \dfrac{10Ps}{B}(R_w R_b + R_e R_b + R_e R_w)}{IR_w R_b - \dfrac{10Ps}{B}(R_w R_b)}\right] \qquad (14)$$

This expression does not take into account the power required to maintain the magnetic field. If the magnet winding is connected in series with the current path through the pump tube, and if this winding has a resistance R_m, then Eq. 14 becomes

$$\text{Efficiency} = \frac{10Ps}{BI} \times \left[\frac{IR_w R_b - \dfrac{10Ps}{B}(R_w R_b + R_e R_b + R_e R_w)}{IR_w R_b - \dfrac{10Ps}{B}(R_w R_b) + IR_m(R_w + R_b)}\right] \qquad (15)$$

It should be noted that the pressure, P, used in the above relations is the pressure which would be developed if there were no hydraulic loss in the tube. The actual pressure, of course, will be less by the pressure drop due to pump-tube impedance. Figure 7 shows how the efficiency of a pump varies with current input. As one would expect from the form of Eq. 14, there is an optimum value of current above which the efficiency decreases because of rising I^2R losses in the tube and magnet.

Types of Pumps

Pumps of several different cross-section geometries have been constructed and tested. These ranged in size from 0.25 × 3.0-in. to 3.0 × 3.0-in. rectangular sections, and one 4.0-in.-diameter circular section. Nichrome, inconel, and stainless steel (347) have been used for the pump tubes. In some cases, the thin-walled tube was fabricated from sheet and then seam welded; in others, thin-walled seamless tubing was used. Wall thickness varied from 0.025 to 0.062 in.

The table on the next page lists the salient data for various pumps.

Method of Testing

For testing purposes, each pump section formed part of a closed loop of either 2.0- or 4.0-in. pipe in which were mounted a throttling valve, flow meters (both orifice and electromagnetic), pressure gages, expansion tank, and Calrod heaters. The expansion tank was located adjacent to the inlet side of the pump. The fluid power developed was taken as being 0.435 × flow rate × Δp watts, where the flow rate is in gallons per minute, and Δp is the difference between the gage readings on each side of the pump.

The systems were filled with sodium-potassium alloy of either the 44% K or the eutectic (78% K) composition. After the systems were filled, the alloy was heated to 350° C for several hours

to ensure wetting of the pump-tube walls. During this procedure, the electrical resistance across the pump tube was observed to decrease by a factor of 5 or more, but when the walls were once thoroughly wetted, the resistance continued to remain low (see the table on this page).

During part of the work, the magnetic field for the pumps was supplied by an electromagnet which was excited by a separate d-c supply. For some of the pumps, however, the magnets were excited by the same current which passed through the pump tube so that the electrical connection was equivalent to that of a series d-c motor.

Construction of Pumps

Since the efficiency of the pump depends to a considerable extent on the magnitude of the power required to maintain the magnetic field, considerable effort was expended in constructing magnets with low power dissipation. The field windings for the series-connected pumps were copper castings machined to fit over the Armco iron pole pieces with close clearance. Pure mica sheets were the only insulation used. When currents greater than 10,000 amperes were required to drive the pumps, the field winding consisted of two turns of 4×4-in. cross section copper or 6×6-in. copper as in the case of the pump magnet for the Experimental Breeder Reactor (Fig. 2).

The leads connecting the pump tube and the magnet winding were of copper bar stock of the same thickness as the pump tube. When these leads were in series with the magnet winding, they contributed an additional turn to it and so aided in raising the efficiency of the system. All current-carrying joints were bolted and soldered. The resistance of these magnet windings was about 6×10^{-6} ohms at 50° C.

The insulation problem is practically nonexistent in these pumps because the maximum voltage drop across the windings is less than 1.0 volt. This is a very desirable feature of this type of pump inasmuch as high temperatures and radiation damage do not affect pump operation. The pumps can be run continuously with liquid temperatures in excess of 400° C without the necessity of any forced cooling.

Sources of Current

The large direct current required for operation of these pumps was supplied initially by a selenium rectifier, capable of continuous operation at currents up to 6,000 amperes. When it became desirable to operate at still larger currents, a homopolar generator was constructed. This machine (*6*) was made very compact by employing NaK alloy as a liquid brush. It was capable of supplying currents in excess of 20,000 amperes at 0.5 volt at efficiencies of

FIG. 8. Homopolar generator (right) and electromagnetic pump (left)

Data for Pumps of Different Cross-Section Geometries

*Pump section** (*in.*)	*Copper area in contact with tube wall* (*in.*)	*Magnet poles* (*in.*)	*Tube resistance, empty, 50° C* (*ohms* $\times 10^6$)	*Tube resistance, filled with NaK 50° C* (*ohms* $\times 10^6$)
$0.25 \times 3.0 \times 6$	0.25×1.0	5.0 dia.	320	90
$0.25 \times 3.0 \times 30$	0.25×5.0	4.0×10	200	30
$0.50 \times 2.75 \times 6$	0.50×1.0	5.0 dia.	300	50
$1.0 \times 5.50 \times 12$	1.0×1.0	4×10	660	42
$1.125 \times 2.50 \times 20$	1.125×5.5	4×10	55	6.0
$1.50 \times 2.00 \times 30$	1.50×5.0	4×10	125	7.2
$1.75 \times 1.75 \times 30$	1.75×5.0	4×10	100	4.25
$3.0 \times 3.0 \times 20$	3.0×4.0	4×10	235	3.2
4.0 dia. $\times$ 20	3.0×4.0	4×10	225	3.3
$1.50(\text{av}) \times 3.5 \times 36$†	1.80–1.43×7.5‡	4×15	290	4.2

* Pumps are referred to in text in terms of dimension parallel to magnetic field direction. In table, first dimension is parallel to field direction, second is parallel to current flow, and third is length of thin-wall section.

† This pump used in Experimental Breeder Reactor.

‡ Tube has taper of 1.80–1.43 in. to provide magnetic field compensation.

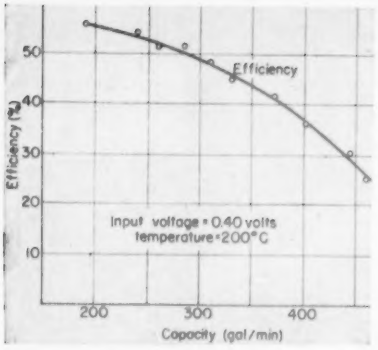

FIG. 9. Characteristics of 1.75-in. pump

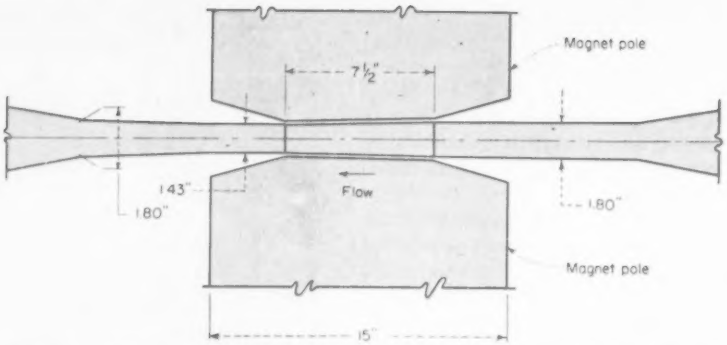

FIG. 10. Tube and magnet-pole arrangement for Experimental Breeder Reactor pump

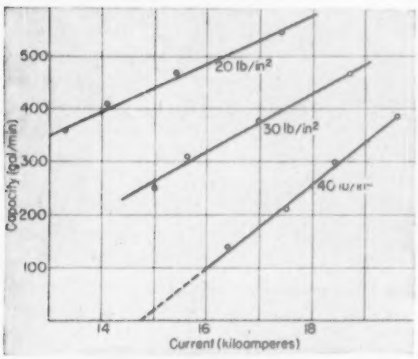

FIG. 11. Variation of capacity with current for EBR pump at 250° C

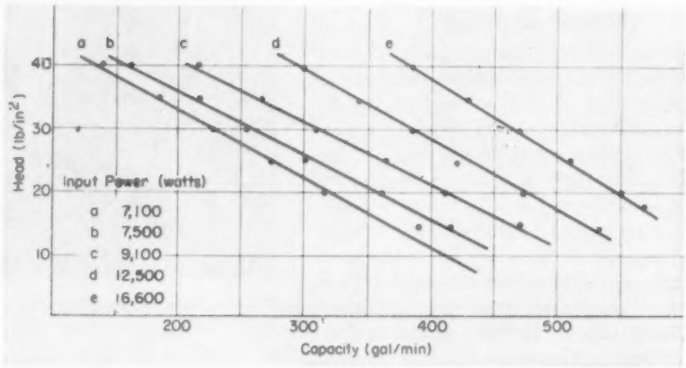

FIG. 12. Characteristics of EBR pump at 250° C

50–65%. This is considerably higher efficiency than can be obtained from rectifiers supplying power at the low voltage required in this application. A homopolar generator connected to an electromagnetic pump is shown in Fig. 8.

The magnitude of the current supplied to the pumps was either read directly with the aid of a standard 10,000-ampere shunt, or in the case of larger currents, by a device operating on the field surrounding one of the bus bars. This consisted of a motor-driven iron block which rotated in the field of a bus bar and adjacent to a fixed pickup coil. The induced emf in the coil was fed to an electronic voltmeter. The arrangement was calibrated against the standard shunt by loading it up to 20,000 amperes for short intervals.

Optimum Field Intensity

From Eq. 12 it is seen that there will be an optimum value of magnetic field intensity for which the flow will be a maximum at a given value of current and pressure. This effect is due to the by-passing of current, which becomes more pronounced as the counter-emf increases with increasing field. Increasing the length of the magnet poles extends the paths which current must follow in the liquid to cross the tube beyond the field region.

A series of tests was made to determine the improvement in capacity and efficiency which resulted from an increase of pole length. It was found, for instance, that with the 1.75-in. pump tube, an increase of pole length from 5 to 8 in. produced a 15% gain in efficiency. When the pole length was then further extended to 10 in., the increase in efficiency was only 3% additional. A further extension beyond the region in which current was crossing through the liquid had a detrimental effect. This is due to the retarding effect of the induced current developed in the moving liquid as it enters the magnetic field. Since these induced currents flow in a direction opposite to that of the current supplied from the external source, they exert a braking effect on the motion of the liquid. To minimize this, the magnetic field intensity should be made to decrease gradually on both the upstream and downstream sides of the region in which current is traversing the liquid. In the ideal case, the falling off of field intensity would coincide with the falling off of current density due to the fringing of current in the liquid beyond the region where the bars are attached to the pump-tube wall. This distribution in field intensity may be accomplished by proper tapering of the magnet poles.

Figure 9 shows a typical relation between efficiency and capacity when the input voltage is kept constant. The efficiency falls off as the rising counter-emf reduces the effective current through the liquid, with a resulting decrease in developed pressure and power output.

The magnetic field due to the flow of current across the pump tube dis-

torts the field between the magnet poles by introducing a component which increases the resultant field intensity on the upstream end and lowers it on the downstream end. This inhomogeneity in the field along the length of the tube produces a greater counter-emf in the moving liquid on the upstream end than on the downstream side with the result that the current distribution along the length of the tube is not uniform but is increased downstream and reduced toward the upstream end. Thus the resultant current and field distribution is such that the regions of highest current density are in regions of lowest field intensity.

The net result of this effect is to lower the pumping capacity. The magnitude of the effect will increase as the current density and counter-emf in the liquid increase. For pumps of large cross section operating at relatively low currents, the effect is hardly noticeable, but for high rates of flow and accompanying large current densities, field compensation should be provided. This may be accomplished in two ways.

In the first method, as in motor design practice, the field of the current in the liquid may be cancelled by an opposing field provided by an equal current flowing in the opposite direction in a conductor which is located above and/or below the pump tube. This is most conveniently accomplished by arranging the bus bars which are attached to the tube in such a manner that current enters the tube from one side and is returned from the other side through a bus bar adjacent to the tube.

The second method is to taper the poles so that the magnetic gap is wider toward the upstream end. In addition, the tube cross section may be tapered, so that the velocity of the liquid increases in the downstream direction at a rate such that the counter-emf in the liquid remains constant as the liquid traverses the region between the poles.

The first method of compensation has the disadvantage that the magnetic gap must be considerably increased to provide room for the compensating bus bars. In addition, the geometry of the whole assembly becomes much more complicated.

The disadvantages of the second method lie in increased tube fabrication difficulties and in the increased hydraulic losses in a tapered tube.

The optimum cross section geometry is determined by considerations of required head and capacity, available current supply, magnet power, and current by-passing losses. Since the by-passing loss increases with counter-emf, and since, from Eq. 7, $E_s = BQ/10^8 s$, large values of s will minimize the by-pass current losses. On the other hand, increasing s requires an increase in current and magnet power.

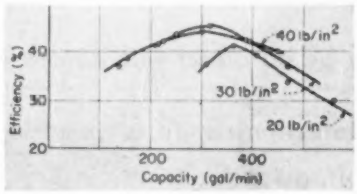

FIG. 13. Efficiency as a function of capacity for EBR pump at 250° C

EBR Pump

The pump constructed for the Experimental Breeder Reactor (EBR) employs a nichrome tube of 0.025-in. wall thickness. The tube tapers from 1.80 to 1.43 in. in the field direction as it passes between the magnet poles. Its dimension in the current direction varies from 2.90 to 3.4 in. The magnetic gap also is tapered from a width of 1.86 in. at the entrance end to 1.50 in. at the exit end (Fig. 10). The pole pieces are 4×15 in. The magnet winding consists of two turns of 6×6-in. cross section copper which are connected in series with the current path through the pump tube. The tube and field structure are insulated from the core by means of mica sheets.

The whole assembly is mounted inside a welded stainless-steel enclosure which will retain the liquid metal in the event of failure of the pump tube. The current leads are brought into the enclosure through stainless-steel bellows which introduce resistance into the short circuit path formed by the enclosure. The normal by-pass current through the bellows is 200 amperes. The bellows are sealed to the 5-in.-diameter copper leads with a pressure fitting. The enclosure is filled with helium at 5 lb/in². Leakage of liquid metal into the enclosure can be detected by the short-circuiting of an insulated probe which is connected to an alarm circuit.

Figure 11 shows the capacity-current characteristics of this pump.

Figure 12 shows the head-capacity characteristics with NaK alloy at 250° C. Each curve shown corresponds to a particular value of power input.

The pump is located in the primary coolant loop of the reactor. It is in parallel with a mechanical pump and arranged so that either pump may be used. The pump is placed in a shielded cell with heavy copper bars passing through the cell wall to a rectifier which is placed externally for ease of servicing. The rectifier is a water-cooled copper sulfide-magnesium unit capable of supplying 20,000 amperes at 1 volt.

In using pumps of this type, it is very important to provide adequate means for venting gas from the system and avoiding situations in which gas can be fed into the moving liquid by turbulence in tanks, etc. The effect of gas bubbles is to increase the electrical resistance through the liquid with resultant loss of flow and efficiency. Difficulty of this kind was experienced in the Idaho installation until baffles were introduced into a tank which was located on the entrance side of the pump.

Figure 13 shows the variation of efficiency with capacity of the pump when operating at 250° C. The over-all efficiency of the d-c type pump involves the efficiency of the current supply system. If this is a rectifier, then for an input voltage of the order of 1 volt an efficiency of 40% is obtainable. This means that the over-all efficiency of the system would be around 20–25%. If a homopolar generator is used, a higher efficiency can be obtained since a generator efficiency of 80%, or more, is possible.

The disadvantage of the d-c electromagnetic pump is the large direct-current requirement.

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Pulsed G-M Tube Operation

These advantages may be obtained with pulsed G-M tube operation:

- **Radiation-intensity detection range is extended by pulse triggering**
- **Strong pulse overvoltage permits minimum dependence on voltage stability**
- **Counting-rate measurement provides a linear-detector-response function**
- **Nonlinear performance may be measured with tube current or count-pulse amplitude**

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THE PRINCIPAL purpose of this article is to illustrate some important aspects of pulsed G-M tube operation.

Two types of pulsed operation will be described. In one of these, the tube is made continuously sensitive to ionizing radiation by energizing it from a d-c source within the G-M plateau region. Upon this voltage there is superimposed a narrow pulse voltage of uniform recurrence rate. In the second mode of operation, the tube is gated by means of a narrow periodic trigger pulse, and it is thereby made responsive to radiation only during the brief interval defined by the pulse duration.

Operation of a G-M tube from a recurrent-pulse voltage increases the range of response to radiation intensity over that obtained with a conventional source of constant potential (*1*, *2*). It may also raise the energy of each counting pulse, as was done by Rossi (*3*) to permit transmission over a long cable. An inherent characteristic of the tube, when operated at constant potential, is the intrinsic time for a discharge to become deionized following a counting event. This condition sets an upper limit to the normal counting-rate capacity.

Continuous Operation

The simplest transition to pulse operation from conventional methods lies in the addition of a recurrent pulse voltage to the d-c G-M voltage region. By this means, the effective quenching action is improved and the useful response is extended to stronger fields of radiation.

With d-c operation alone in strong radiation fields, the typical tube performance is of the form shown in the Stever diagram (*4*, *5*) of Fig. 1. Here the significance of the deadtime and of the recovery time is clearly evident. During the deadtime interval following

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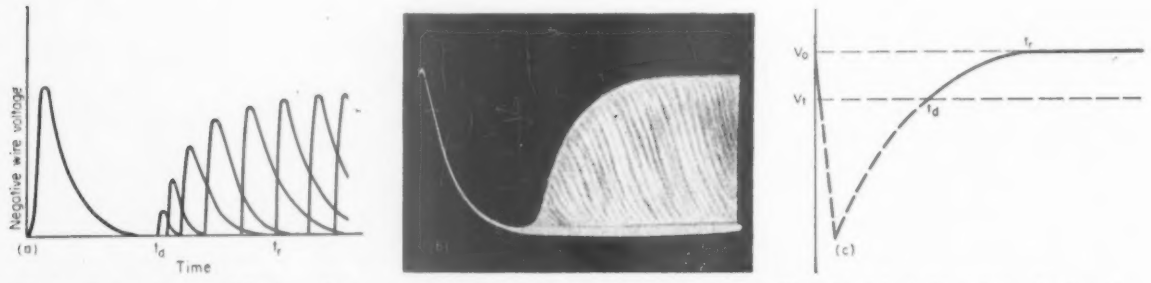
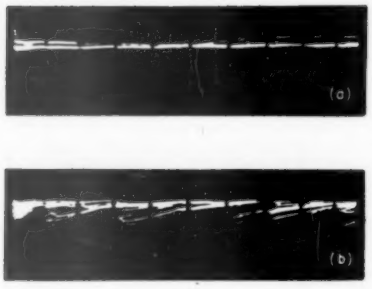


FIG. 1. Stever diagram: (a) schematic representation of deadtime pattern indicating deadtime, t_d, at foot of envelope of pulses triggered during recovery interval from t_d to t_r; (b) deadtime pattern photographed on triggered sweep; (c) variation of electric field at the anode surface during period of ion-sheath transport from anode to cathode



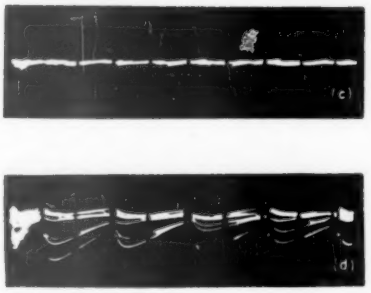


FIG. 2. Oscillograms of G-M tube performance for d-c and for d-c-plus-recurrent-pulse operation

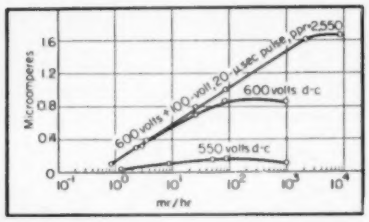


FIG. 3. Tube current vs. radiation intensity

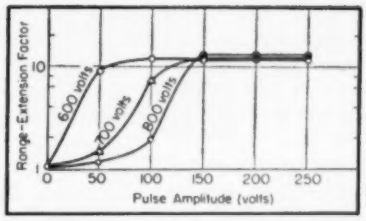


FIG. 4. Range-extension factor vs pulse amplitude with pulse rate of 2,550 pps

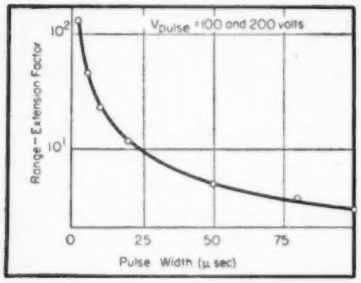


FIG. 5. Range-extension factor vs pulse width with pulse rate of 2,550 pps

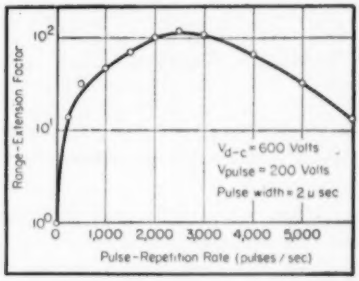


FIG. 6. Range-extension factor vs pulse-repetition rate

a count, the tube is paralyzed as a result of field reduction in the vicinity of the wire arising from the relative immobility of the positive ion sheath during the electron transit time; this condition renders the tube incapable of further response to ionizing radiation during this period.

The effect of exposure to various radiation levels on G-M tube pulse amplitude is shown in Fig. 2. The oscillograms were made with a 2-in. mica-window G-M tube with 100-μsec markers. Tube voltage for all four cases was 600 volts, d-c. Excitation was 10 and 200 mr/hr and 100 r/hr for Figs. 2a, 2b, and 2c, respectively. Figure 2c shows the tube in a state of saturation; it is virtually blocked by the strong radiation exposure. Figure 2d illustrates the kind of recovery that takes place due to superimposing a narrow periodic pulse voltage upon the d-c tube voltage. Pulse width was 5 μsec; pulse rate, 5,000 pps; and pulse amplitude, 800 volts.

A similar set of characteristics showing G-M tube current as a function of radiation intensity level appears in Fig. 3. The tube used was a type GC120N with a 2-in. mica end window; threshold is 560 volts with 40-cm Ne and 1-mm Cl_2. Adding a periodic pulse (width = 20 μsec, rate = 2,550 pps) to the tube voltage extends the linear region without affecting the performance below saturation for the d-c case alone.

Combined pulse-plus-d-c operation raises the radiation level at which saturation occurs. This, for convenience, is defined as the limiting radiation level beyond which the slope of the detector-current response characteristic changes sign.

The quality of performance obtainable from superimposed pulse operation is a function of the tube deadtime, the operating voltage, pulse amplitude, pulse width, and repetition rate. These functions are illustrated in Figs. 4 to 6, using the GC120N tube. In these figures, the range extension factor used as the ordinates is defined as the ratio of the saturation radiation fields for combined pulse and d-c operation to d-c operation alone.

Counting Rate vs Dosage

Referring again to Fig. 2d, it becomes evident that the tube recovers in the period immediately following an injected pulse, so that quenching is effectively improved. In the presence of strong radiation fields, such performance would also indicate a convergence of the recovered G-M tube counting rate towards the pulse repetition rate. This is brought out in Fig. 7 with the GC120N tube operating at 600 volts, d-c. Counting-rate saturation occurs at 12 mr/hr for the d-c case (curve A). With combined 90-volt pulse and d-c operation at a discrimination level of 0.5 volt, and for the same pulse-amplitude detection-sensitivity condition as for curve A, curve B is obtained. The maximum counting rate is seen to converge in a nonlinear manner towards the pulsing frequency. Such nonlinear performance is generally unsuitable for measurement purposes. By raising the amplitude discrimination level to 3.0 volts, curve C is obtained. This converges towards the pulsing frequency at 560 mr/hr in a manner more compatible with quantitative measurement requirements.

The amplitude distribution of count pulses as a function of counting rate is illustrated in Fig. 8. Curve A was obtained with a type-120N tube operating at 600 volts, d-c; curve B was obtained with a 90-volt pulse at a rate of 2,000 pps. For d-c operation, total pulse > 0.5 volt = 114,000 cpm; for pulsed operation, total pulse > 0.5 volt = 98,400 cpm.

In the example (Fig. 7) of counting-rate detection, the radiation-intensity range has been increased by a factor of approximately 46 by the introduction of a recurrent-pulse voltage upon the G-M d-c voltage region. As will become apparent, the limiting pulse-

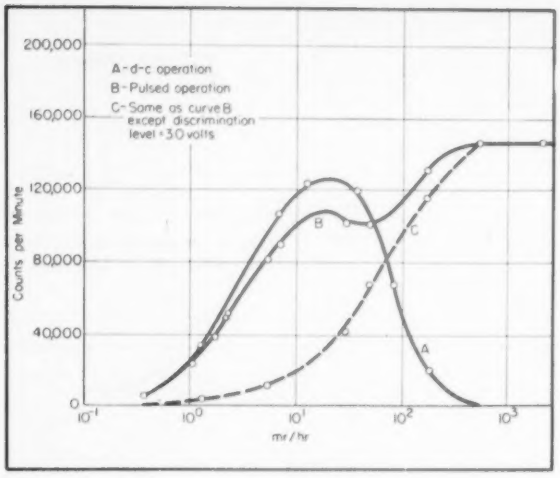


FIG. 7. Counting rate vs radiation intensity

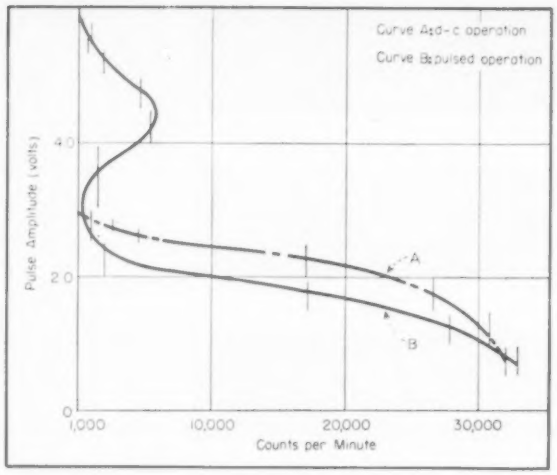


FIG. 8. Pulse-amplitude distribution vs counting rate with dosage = 10 mr/hr

repetition rate (PRR), and therefore the highest linear counting rate that may be extracted by this means, is related to the tube deadtime.

Tube Current vs Dosage

An analogous set of curves representing tube current as a function of radiation intensity level is shown in Fig. 3. For d-c operation alone, the current curves corresponding to two levels of d-c operation exhibit saturation at about 90 mr/hr. Above this level of exposure, current decreases with further increases in radiation intensity.

When the d-c voltage is supplemented with a recurrent-pulse voltage, the tube-current response becomes linear out to approximately 3,000 mr/hr. Beyond this point saturation effects again become noticeable. The superimposed pulse will be observed to have straightened out the current-radiation field characteristic in the region wherein curvature would have otherwise occurred for d-c operation alone. This was attained without altering the performance at lower intensity levels in the region within the normal measurement capabilities of the tube. For the example chosen, the use of combined pulsing and d-c operation has extended the useful radiation-intensity capacity of the tube by a factor of 33.

Pulse Parameters

The effect upon tube performance arising from the introduction of a recurrent voltage pulse into a conventional G-M tube circuit is dependent upon all of the features that characterize the pulse. The repetition rate, pulse width, pulse amplitude, and pulse shape are each separate and controlling entities. Effects of these parameters upon the performance are shown in Figs. 4 to 6. A rectangular pulse was employed in all tests. The effect of pulse shape upon performance has not been separately investigated as this, it was felt, may be largely inferred from the study of pulse width.

Pulse amplitude. Figure 4 shows the influence of pulse amplitude upon the radiation-intensity-versus-current characteristic of a G-M tube for three levels of d-c voltage. Each d-c voltage level falls within the G-M plateau region. The range-extension factor from this figure is independent of where on the plateau the d-c operating point is chosen. Essentially identical performance is obtained from each of the three operating states shown, provided, however, that the pulse amplitude used is not less than 20% of the d-c voltage employed. The use of a larger pulse amplitude does not further extend the saturation capabilities as the curves simply level off. With narrow pulses, the combined pulse plus d-c voltage may rise far beyond the G-M plateau boundaries without adversely affecting the performance. Spurious counts are inhibited by the deliberate choice of a pulse width much less than the tube deadtime and of periodicity approximating the reciprocal of the deadtime.

Pulse width. To improve tube quenching, the pulse width chosen must be much less than the tube deadtime. This is brought out in Fig. 5 from which it may be seen that the range-extension factor varies inversely as the pulse width. Continued improvement in operation down to the narrowest pulse employed, namely, about ½-μsec duration, is evident from this figure. The ultimate extension in range, as will be brought out in the discussion on triggered G-M tube operation, will be shown to reside in the extension factor t_d/w. Here t_d is the G-M tube deadtime interval, and w is the width of the injected pulse.

For the above tests, the injected pulse duration was large by comparison with the electron transit time. As long as this condition is maintained, the conclusion drawn therefrom, namely, improvement in range extension with reduction in pulse width, follows. If the pulse were narrowed sufficiently so as either to approximate or become less than the electron transit time, the counter-tube behavior would be altered markedly in accordance with transit-time theory. One of the effects to be anticipated as such conditions of operation are approached would be a reduction in influence of the pulse upon the tube performance. As the electron transit time is of the order of 10^{-8} sec, this magnitude serves to establish a possible lower limit to the pulse duration that may be employed. This matter merits further investigation.

With reference to Figs. 2, 3, and 4, the injected periodic-pulse voltage may be seen to assume the role of a catalyst by speeding recovery from each discharge without materially affecting the count-pulse energy.

Pulse-repetition rate. The influence of repetition rate upon range ex-

tension is illustrated in Fig. 6. An optimum well-defined rate occurs, for the tube illustrated, at a repetition rate of 2,500 cps. This period agrees identically with the tube deadtime interval, namely, 400 μsec, which was verified by means of a Stever oscilloscope pattern.

This behavior is understandable when it is observed that a pulse-repetition period longer than the deadtime will result in the tube's saturating and blocking up between pulse intervals. Also, when the pulse-repetition period is shorter than the deadtime interval, complete recovery between pulsing events will not occur, with a consequent lowering in range and in the maximum G-M tube current. Maximum circuit-current conditions and a capacity for covering the greatest range of radiation intensity are both, therefore, satisfied by a repetition period equal to tube deadtime. For direct-counting applications, these criteria also fix the maximum counting rate as the reciprocal of the deadtime interval.

Intermittent Operation

If the high voltage furnished to a G-M tube is applied for brief intervals at a periodic rate, the tube is rendered sensitive to radiation only while the voltage is impressed upon it. The measured counting rate, when triggered in this manner, will be a fraction of the count that would have been measured had the voltage been constantly applied to the tube. The ratio between both counting rates may, for convenience, be called the scale factor, *SF*, or

$$SF = \frac{\text{Counting rate for continuous (d-c) operation}}{\text{Counting rate for intermittent (triggered) operation}} \quad (1)$$

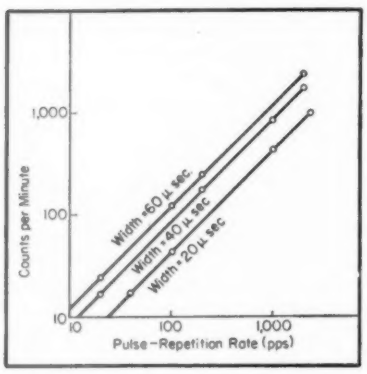


FIG. 9. Pulsed G-M tube counting characteristic

According to this definition, $SF > 1$.

The relationship between trigger-pulse parameters, detectable counting rate, and the scale factor *SF* is illustrated in the experimental curves of Figs. 9 and 10. In the case of Fig. 9, the tube was operated at 600 volts, d-c, with a 400-volt pulse amplitude; the counting rate with d-c operation was 15,722 cpm. The results of Fig. 10 were obtained with a type-BS1 tube operated at 570 volts, d-c, with a 90-volt pulse and a 997-pps rate. For pulse widths considerably less than the tube deadtime, *SF* is equal to the reciprocal of the duty factor

$$SF = 1/wf \quad (2)$$

where w is the pulse width and f is the repetition frequency.

Equation 1 may now be given as

$$\Delta f = F(R)wf \quad (3)$$

where Δf is the measured counting rate and $F(R)$ is the maximum counting rate (corresponding to continuous sensitivity) and is a linear function of the radiation intensity level, R.

From the foregoing, it is clear that by pulse trigger operation, the total potential counting rate is lowered at the tube by the duty factor wf. This feature, for all practical purposes, removes the tube's saturation limitations in high-radiation fields.

Choice of Trigger-Pulse Rate

If the pulse-rate frequency, f, is held constant for a given sensitivity scale range, Eq. 3 may be rewritten in the form

$$\Delta f/f = F(R)w \quad (4)$$

From this expression it is apparent that Δf will approach f as a limit. The left-hand member will, therefore, approach unity, regardless of the choice that is made in f. The range-extension capabilities under these circumstances are then seen to reside entirely in the choice made for pulse width, w. The PRR value, f, therefore becomes a matter to be determined by the statistical accuracy requirements of the measurement data, rather than one of range extension. Furthermore, to provide for complete recovery of the counter-tube discharge between successive counts, f must not exceed the rate prescribed by the reciprocal of the deionization or recovery time. Otherwise, the discharge will be reignited by the triggering pulse and it will be self-sustained at the trigger-pulse rate.

Recent experiments with halogen-filled tubes, of the type BS1 variety, have shown that normal counting operation occurs for triggering intervals reduced to less than 10% of the G-M tube deadtime. This very interesting phenomenon is worthy of further investigation.

Choice of Pulse Width

To examine further the influence of trigger pulse width, w, upon the range extension capabilities, it is necessary to rewrite the counting-rate function $F(R)$ in Eq. 4. $F(R)$ is the counting rate which the tube is capable of if operated normally so as to be continuously sensitive to radiation. In the limit, the counting-rate capability would then be determined by the reciprocal of the tube deadtime period, or

$$F(R_{Max}) = 1/t_d \quad (5)$$

With the insertion of this limiting value in Eq. 4, this becomes

$$\Delta f/f = w/t_d \quad (6)$$

With the choice of w/t_d less than unity and the observation that this quantity is the fraction of maximum counting-rate output available from triggered operation, which corresponds to a radiation field R_{Max}, then the system is capable of normal performance up to a factor t_d/w of this amount before saturation will occur. This will be marked by the left-hand term of the equation becoming unity.

If R_{Max} is then the strongest radiation field that a counter tube is capable of being used in with d-c operation, due to deadtime, t_d, limitations, and if R_p is the strongest radiation field which this same tube is capable of being used in under triggered conditions of opera-

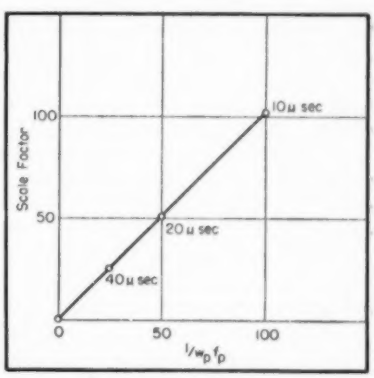


FIG. 10. Scale factor vs reciprocal-duty factor

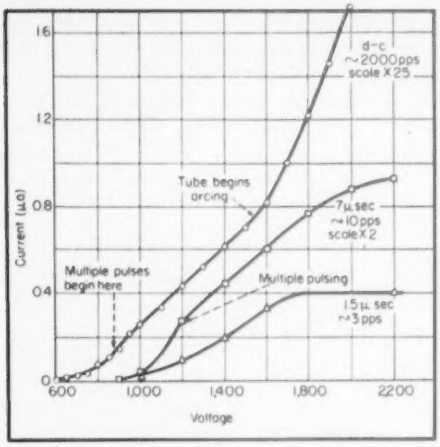


FIG. 11. Tube current vs voltage for high overvoltages

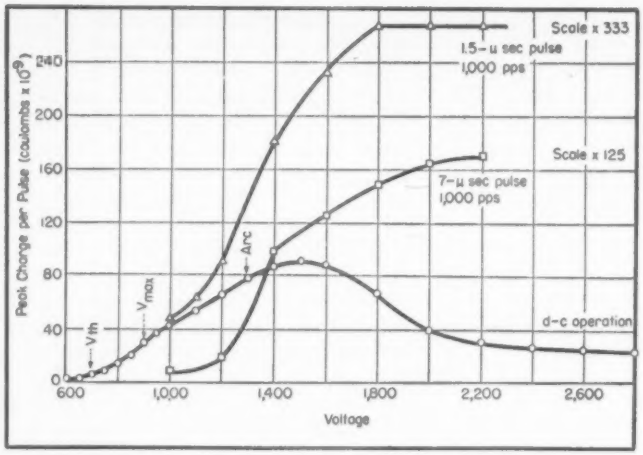


FIG. 12. Charge per pulse vs voltage for high overvoltages

tion, using a pulse of width w, it follows that

$$R_p = R_{Max} t_d / w \qquad (7)$$

The range-extension factor is clearly seen to be the ratio of the tube dead-time to the triggering-pulse width. Equation 7 may also be seen to generally describe the behavior of the experimental curve of Fig. 5 for the combined d-c and FRR operation.

Large-Overvoltage Operation

The use of a narrow trigger pulse for energizing a G-M tube precludes gaseous discharge except by pulse coincidence with an external ionizing agency. This feature extends counter operation far beyond the usual limitations imposed by the Geiger plateau region, to very high overvoltages. The ultimate useful limitation of the impressed voltage under these circumstances is determined by the insulation capabilities of the tube terminals rather than by restrictions imposed by properties of the gaseous vehicle itself. Such counter operation projected into the realm of strong transient voltage fields introduces a number of potentially useful performance phenomena.

Typical performance curves obtained for a type BS1, halogen-filled G-M tube (Figs. 11 and 12) disclose several items of interest:

1. Normal count-detection operation was limited by the voltage range of the available pulsing source rather than by the tube characteristics.

2. The counting threshold for triggered operation is several hundred volts higher than for the corresponding d-c case. This condition may be brought about by a reduction in energy per count pulse resulting from an acute reduction of the spread of discharge—the latter being restrained by the brief duration of the impressed voltage.

3. The charge per count pulse increases at first linearly with overvoltage. But for operating voltages approximating three times the normal d-c G-M threshold value and beyond, the charge per pulse levels off and becomes virtually independent of the tube voltage. This operation is best realized with the narrowest pulse used (1.5 μsec). The pulse charge is seen to grow in size at first. It finally levels off after reaching a magnitude approximately 10^4 times greater than is obtainable from conventional d-c operation in the Geiger region.

A closer examination of the character of the discharge, in the region where leveling off of the pulse size occurs, has shown the light output of the tube to increase proportionately with voltage while the ionization current remains substantially constant in value. Over this realm of operation, increases in tube voltage seem to favor the excitation of light radiation without influencing the total ionization. Fluctuations in tube overvoltage are thereby converted to corresponding changes in light energy with little or no influence upon the ionization current. Although the curves (Figs. 11 and 12) are shown as taken out to 2,200 volts, additional tests have shown that the tube current was still independent of impressed voltage at 5,300 volts.

The removal of an upper voltage limit in the useful employment of counter tubes, aside from insulation requirements, permits high voltages to be utilized with the accompanying benefit of dealing with stronger charges from each counter pulse. If the triggering pulse is narrowed sufficiently, the tube voltage may be raised so that operation may be confined to the region wherein the tube current becomes largely independent of the amplitude of the impressed voltage. This characteristic may prove useful in instrument designs by either reducing or completely eliminating dependence of instrument calibration upon tube over-voltage. It also furnishes a basis for dispensing with the need for high-voltage stabilizers for counter tubes.

Since triggered operation enables only one measurement count to occur for each triggering pulse, it becomes obvious that spurious counts are inhibited from developing.

Choice of Gas Filling

The trigger-pulsing technique is not critical with regard to the choice of filling gas used. Any pure gas, whether used alone or in combination with other gases and whether of monatomic or of polyatomic structure, will be found capable of performing as a radiation counter. This latitude in the selection of gas fillings further enhances the desirability and potential usefulness of triggered operation.

Counter tubes of selfquench or non-selfquench characteristics may be used. Selfquench tubes offer the advantage of fast operation combined with mini-

mum circuit complications arising from their inherently rapid deionization properties. The choice of a filling medium that does not deteriorate with use, such as is obtainable from the use of halogen gases in combination with neon or argon (*6*), removes any upper limiting restriction on the overvoltage that may be beneficially employed. Operation may be thereby extended into the realm of constant-ionization currents, with ensuing advantages of maximum energy per tube count and greatest operational immunity from tube-voltage amplitude variations.

The use of an organic or polyatomic vapor for the filling medium imposes a practical upper limit upon the overvoltage that may be used, as determined by reasonable tube-life expectations. This limitation arises by virtue of the dissociation that occurs in such a gas, which varies directly with the intensity of the discharge current.

Counters filled with a pure gas such as neon or argon are of the nonself-quenching variety but may also be made to serve as satisfactory triggered counters. Tubes of this type are desirable in view of the filling-gas simplicity and the high order of stability that may be readily obtained therefrom. The trigger pulse must not be reapplied, however, until complete deionization has taken place from the previous count. Otherwise the discharge will be sustained at the pulsing rate and the usefulness of the tube as a radiation counter will be destroyed. As this process is slow in such a tube, the triggering rate must also be kept proportionately low. In a typical counter tube filled with either argon or neon gas, for example, it was found necessary for the triggering rate to be kept below 15 pulses/sec for normal counter operation to exist. Faster operation may be acquired and the deionization time may be hastened, however, by the use of a suitable d-c collecting field in combination with the triggering pulse, such as was done by Rossi (*3*).

Design Considerations

Equation 3 is in a form that is useful for design applications. As stated therein, the total available counting rate, $F(R)$, expressed here as a function of the radiation intensity level, is scaled down by factors w (pulse width) and f (pulse-repetition rate) to a detectable rate, Δf. Consequently either or both

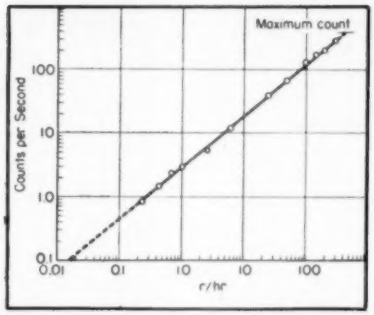


FIG. 13. Counting rate vs radiation intensity

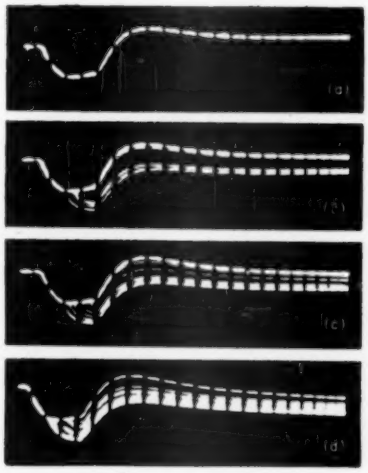


FIG. 14. Trigger-pulse operation of G-M tube with no excitation (a), low (b), medium (c), and high excitation (d)

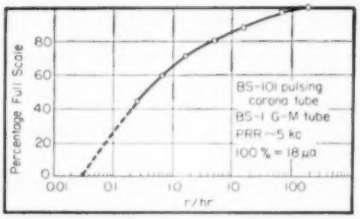


FIG. 15. Tube current vs radiation intensity

of these parameters may be controlled so as to provide the desired amount of scaling. The pulse width is usually most readily kept constant, thus enabling an instrument's scale range to be calibrated by adjustment of the pulse-repetition rate.

Range Linearity

The G-M tube operated from a pulse-triggering source may be made to perform either as a linear or as a nonlinear radiation detector. Pulse-counting operation, as typified by Eq. 3, yields linear performance. Results obtained with this technique are shown in Fig. 13; 4.7-μsec-wide pulses at a rate of 400 pps were detected with a BS2 tube.

Where several decade levels of radiation intensity are required to be covered on one scale range, nonlinear performance and scale compression become essential. This may be achieved in a device utilizing either tube current or count-pulse amplitude as the measuring quantity. In both instances nonlinearity arises by virtue of the interruption of the discharge before it has had sufficient time to spread fully. The spread velocity of the discharge is of the order of 10^6 to 10^7 cm/sec (*5*). This phenomenon yields an accumulation of progressively smaller count pulses in unit time as the quantity $\Delta f/f$ in Eq. 3 approaches unity and as the saturation capabilities of this sort of device are approached. Discharges occurring near the leading edge of the trigger pulse will have the greatest time to spread and will thereby contribute a larger amplitude than those that are formed near the trailing edge. The gradation in pulse sizes that occur over the triggering-pulse pedestal may be seen in the oscillograms of Fig. 14. Here a type-BS1 tube detected pulses of 800-volt amplitude, 5-μsec width, and 500-pps rate, with no d-c voltage; markers were 1 μsec apart.

Control of the trigger-pulse shape furnishes a means for attaining a desired amount of compression. A trigger pulse of long-rise and rapid-decay times will tend towards equalization of the amplitude distribution of count pulses that occur over the triggering interval, with a consequent trend towards linear performance. Conversely, a triggering pulse of rapid-rise and long-decay characteristics may be used to provide scale compression in the vicinity of strong field operation. Figure 15 illustrates a compression characteristic obtained for a triggered G-M tube instrument by utilizing tube-current measurement. This device covers several decades of intensity levels.

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Radiochemistry in Electronics Research

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MANY PROBLEMS of current interest to the electronics industry, as typified by our research program, are concerned with the effect of minute amounts of impurities on the properties of various materials. The electrical behavior of semiconductors, for instance, is greatly dependent upon the presence or absence of extremely small amounts of certain elements. In the case of germanium, used in the construction of the transistor, it is important to analyze quantitatively for one part of impurity in one hundred million parts of germanium. One of the few analytical methods operative in this concentration range is the use of radioactivity, which includes both activation analysis and tracer techniques.

A typical application of the activation method is illustrated here by a study of the contribution of trace amounts of impurities by highly pure graphite boats and crucibles to germanium metal during melting at 1,000° C in a hydrogen atmosphere for 1½ hours.

The use of the tracer technique is illustrated by a study of the segregation of trace amounts of impurities in germanium metal during a directional cooling process, in which the germanium is completely melted and then progressively cooled from one end of the sample to the other. The use of autoradiography in combination with counting methods has proved to be a very valuable research tool.

Fractional distillation and solvent extraction have also been studied here with tracers. These chemical separation processes, besides being of interest in work with germanium, are effective in preparation of radioactive compounds.

* * *

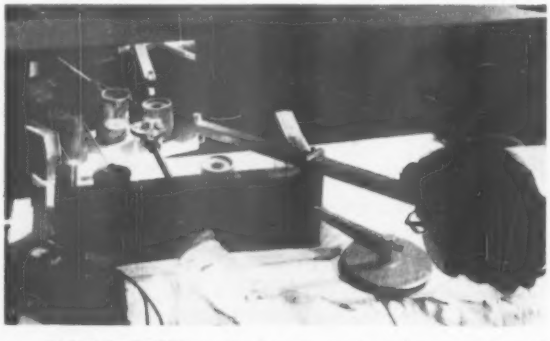
All photographs, including the cover of this issue, are by L. F. Ankersen of Sylvania Electric Products Inc.

ACTIVATION ANALYSIS

1 TO INVESTIGATE transfer of impurities to germanium metal during melting, miniature boat and crucible are irradiated in Brookhaven reactor to activate trace impurities. Container is ¾ × 2⅛ in. to fit reactor conveyor system. To aid radiochemical analysis, nonradioactive graphite was previously ashed and spectrographically analyzed

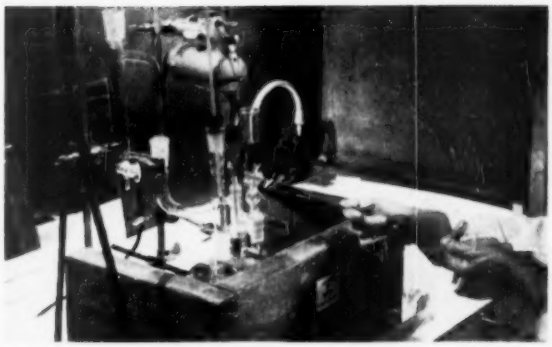
4 CHEMICAL SEPARATIONS and standard radiochemical techniques are used to separate impurity elements into different analytical groups and eventually to prepare individual samples in planchets for counting. Radioactive impurities leached from graphite are similarly analyzed

TRACER STUDIES

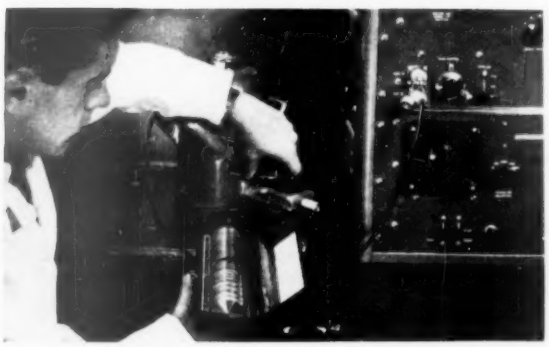


7 RADIOISOTOPES are used as tracers to study segregation of impurities by directional cooling of germanium metal. From aliquots of the isotopes, homogeneous mixtures of impurities and germanium are prepared and subjected to directional cooling. Commercial remote handling device, modified to fit base of pipetting device for support, aids in preparation

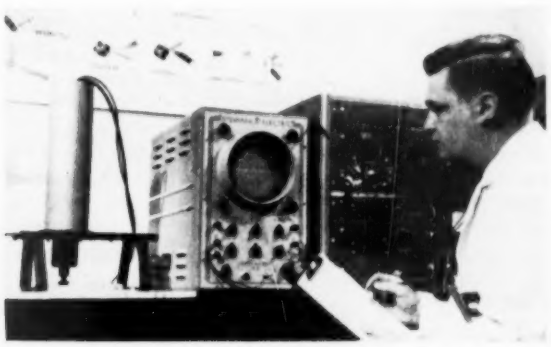
2 LUCITE DRY BOX is used to confine radioactivity during powdering of graphite container and of germanium ingot produced in it. File is used to powder graphite, from which radioactive impurities are leached and then submitted to radiochemical analysis. Tray-shaped paper covering in bottom of box is used to collect filings



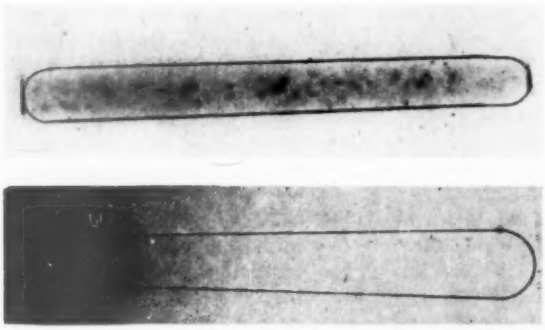
3 POWDERED GERMANIUM is dissolved and converted to germanium tetrachloride. Carrier solutions are added and germanium separated from impurities by distillation. Simple distillation apparatus has additional funnels for carriers and reagents, as well as take-off for tetrachloride. Lead bricks shield stillpot for impurities



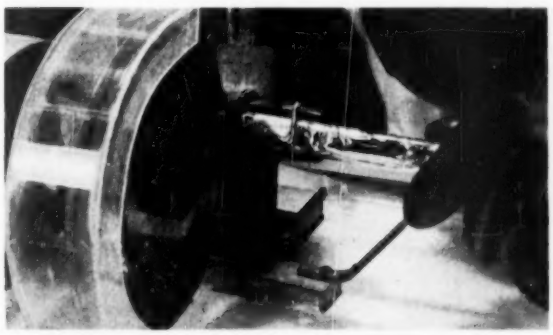
5 MEASUREMENT of impurity activity is accomplished by Geiger counting. Lead shield for counter was modified to reduce background count; new sample holder gives 2-in. lead shielding on all sides of sample. Impurity isotopes are identified by differential decay or absorption methods



6 SCINTILLATION COUNTING is used when activity and energy of radiation make it advisable. Geiger and scintillation results indicate that over 10 elements transfer to the germanium from the graphite containers. Total per cent transfer is in the range of 2–4%



8 GERMANIUM is completely melted, then progressively cooled from one end of sample to the other. Each impurity distributes between the solid and liquid metal in a different ratio. Radioautographs provide visual method of observing distribution. Typical sample ingot is shown before (top) and after (bottom) directional cooling



9 INGOT is sliced with diamond wheel enclosed in dry box to confine activity. Slices of germanium are crushed, mounted for counting. Activity measurements identify impurities, are used to calculate distribution coefficients of impurity elements between liquid and solid germanium. This information is needed to improve purification

The Reactor as a Neutron Source

Neutrons from a reactor are divided into three energy groups: fast, resonance, and thermal. The energy distribution, flux, and spatial distribution of the neutrons in each group is discussed here, and the fission spectrum is given

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NEUTRONS FOR RESEARCH uses can be obtained with a wide energy range by using a reactor as a source. This article is concerned with the characteristics of the three main energy groups of neutrons available from research reactors.*

Fast Neutrons

In considering the neutrons in the pile with respect to their use in research, it is customary to divide them energy-wise into three groups: fast, resonance and thermal. These energy classifications, in spite of their titles, refer rather to the manner in which the neutrons are obtained in the pile than to the nuclear reactions produced by them. The fast neutrons are those produced in fission that have not been moderated by collisions. The resonance neutrons comprise the $1/E$ spectrum produced by moderation and they are usually considered to extend in energy from about 1 Mev down to 1 ev. The thermal neutrons have reached equilibrium with the graphite atoms and possess a Maxwellian distribution of velocities.

In the simple form of finite-pile theory already discussed (*1*), the neutrons were considered as if they were all of thermal energy. This one-group theory led to a simple pile equation whose solution for a rectangular pile showed that the neutrons would have a cosine distribution.

A more accurate form of pile theory must take into account the fact that fission neutrons move an appreciable distance before becoming thermal. In discussing the infinite-pile lattice (evaluation of k_∞), the effect of fast neutrons is treated correctly by inclusion of the factor ϵ for fission caused by fast neutrons, and the factor p for absorption of resonance neutrons. In the case of the finite pile, this additional feature is included in an approximate way by changing L^2 to M^2 to allow for the migration of neutrons during moderation.† Fortunately, the spatial distribution of neutrons of *all* energies in an actual pile is given reasonably well by the solutions of the steady-state pile equation, which is just the one-group pile equation written in a way to include the effect of fast neutrons.

The reason for the similarity in behavior of fast and thermal neutrons is that the thermal-neutron distribution determines the rate of fission in the uranium; as a result, the emission rate of fast fission neutrons will have the same distribution as the thermal neutrons. As the fast neutrons are slowed down, they move over distances of the

Symbols Used

E = neutron energy
f = thermal utilization
j = number of collisions to reduce neutron energy to thermal
k_∞ = reproduction factor for infinite lattice = $\eta\epsilon pf$
L = diffusion length
M^2 = migration area = $L^2 + \tau$
N = number of nuclei per cm^3
n = neutron density (cm^{-3})
nv = neutron flux (cm^{-2} sec^{-1})
p = resonance escape probability
q = neutron slowing-down density
q_0 = slowing-down density at fission energy (about 2 Mev average)
q_t = slowing-down density just above thermal (0.4 ev)
t_0 = life of thermal neutrons in pile
v = neutron velocity
v_0 = most probable velocity in Maxwell distribution
ϵ = fast effect (effect on k of fissions caused by fast neutrons)
ξ = average change in $\ln E$ per collision with moderator atom
η = neutrons produced per thermal neutron absorbed in uranium
Λ = total path length
λ = scattering mean free path
λ_{tr} = transport mean free path
σ_a = absorption cross section
σ_s = scattering cross section
σ_{th} = thermal absorption cross section
τ = neutron age (mean-square slowing-down distance = 6τ)

* From the forthcoming book, "Pile Neutron Research," to be published in February, 1953, by Addison-Wesley Press Inc., Cambridge, Mass.

† In the steady state pile equation $-\frac{\nabla^2 n}{n} = \frac{k_\infty - 1}{M^2}$.

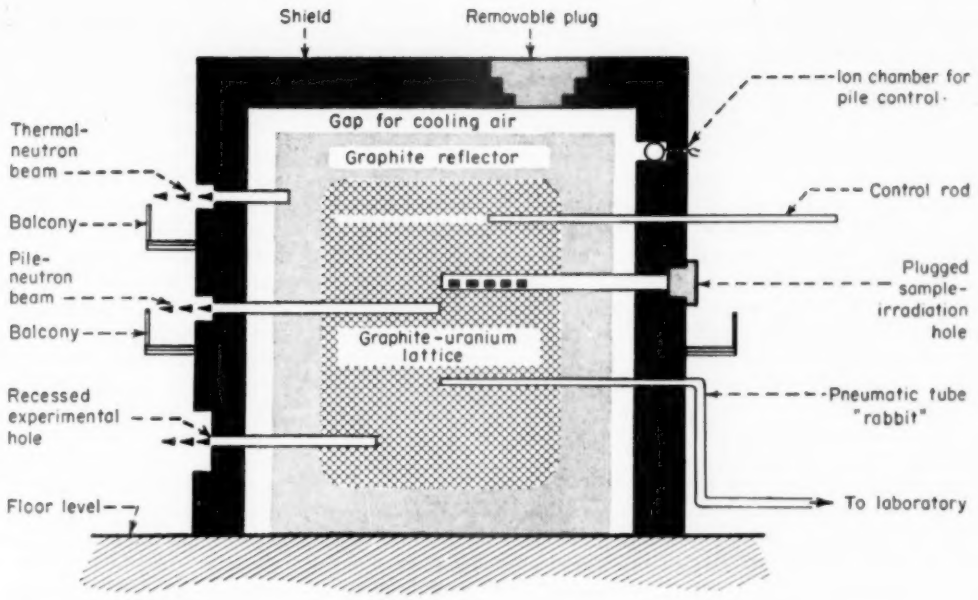

FIG. 1. Cross section of Brookhaven reactor showing features of importance for neutron research

order of $(6\tau)^{1/2}$, the root-mean-square slowing down distance, which for graphite (λ and $\lambda_{tr} \sim 3$ cm, $j \sim 100$) is about 40 cm. Since this distance is small compared with the dimensions of the pile, the moderated neutrons as well as the fast neutrons follow approximately the same distribution as the thermals.

Outside the pile lattice itself, that is, in the *reflector* of the typical research reactor shown in Fig. 1, the distributions of neutrons of various energies differ greatly. In the reflector there is no production of neutrons, hence the flux of neutrons of a particular energy, as we shall see, is determined by the transmission through graphite for that energy.

Even before moderation, the fission neutrons exhibit an extremely wide energy range, from zero to at least 15 Mev. Early work on the upper limit of the energy spectrum indicated that no neutrons were present with energy higher than 6 Mev, but measurements of increased precision have pushed the upper limit much higher.

The most accurate fission spectrum known at the present time is given in Fig. 2. The shape of this spectrum at high energy was determined (*2, 3*) from the range of recoil protons produced in a counter; below 1 Mev, the recoil protons were measured (*4*) in a hydrogen-filled cloud chamber. The energy distributed is reproduced within experimental error by the empirical formula

$$N(E) = \sinh (2E)^{1/2} e^{-E} \qquad (1)$$

where E is in Mev.

This formula is consistent with a model in which the neutrons are emitted, or "boil off," from the highly excited fission fragments after they have separated. According to this model, the observed neutron-energy distribution is that of the evaporated neutrons relative to the fission fragment, modified by the motion of the fragment itself.

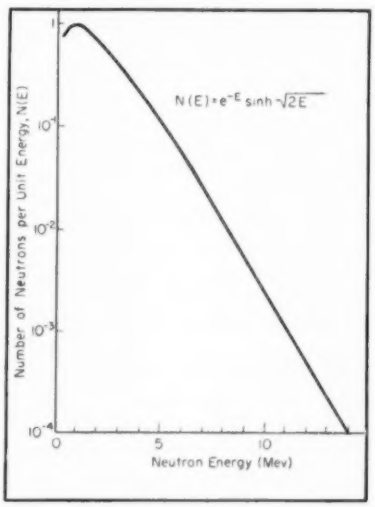

FIG. 2. The fission spectrum

The neutrons of any particular energy in the fission spectrum attain a $1/E$ energy distribution after colliding with the moderator atoms. The actual energy distribution in the moderator will thus be a sum over many $1/E$ distributions, each one extending up to a particular energy in the fission spectrum. Because of the $1/E$ flux distribution, there are many more low-energy neutrons per unit energy than unmoderated fast neutrons in the graphite of the pile. It is difficult to observe the effects of the fast neutrons alone in an experiment inside the pile lattice because of interference by the large flux of moderated neutrons.

It is easy to see that the flux of fast neutrons (of all energies in the fission spectrum) near a uranium fuel rod is of the same order of magnitude as the thermal-neutron flux at the same point. Most of the thermal neutrons hitting the uranium rod will be absorbed in it, and for each one so absorbed, $\eta\epsilon$—or 1.36 neutrons (*1*)—will reappear as fast neutrons. The number of these fast neutrons crossing a square centimeter at the surface of the rod, or the fast flux, will then be of the same order of magnitude as the thermal flux. The intensity of fast neutrons of a particular energy drops off rapidly with distance away from the uranium however, as given by the Gaussian distribution of neutrons from a point.

The distance covered by the neutrons

in slowing down completely to thermal energy is larger than the distance (about 20 cm) between uranium rods; as we have just seen the mean-square distance is about 40 cm. The number of thermal neutrons, on the other hand, actually increases with distance away from the uranium rod because the absorption in the rod lowers the thermal flux for a distance of the order of a scattering mean free path around the rod. As a result of these spatial variations, the ratio of the flux of fast to thermal neutrons will decrease rapidly as a function of distance from the uranium.

Neglecting the finer details of the neutron distribution, e.g. the variation of the order of the spacing of the uranium rods, we thus see that the fast neutrons show approximately the same broad distribution as the thermal neutrons. This distribution, within the lattice, is simply a cosine function for a rectangular pile, or a Bessel function for a cylindrical pile, that goes to zero one extrapolation length beyond the lattice boundary. In the reflector surrounding the pile, the ratio of fast to thermal neutrons will differ markedly from its value within the pile lattice, and furthermore will vary with distance in the reflector. The thermal neutrons will diffuse through the reflector, decreasing exponentially with a period, or "relaxation length," equal to the diffusion length, in accordance with simple diffusion theory.

The fast neutrons will decrease with distance in a much more complicated manner, based on a sum of Gaussian slowing-down distributions. The decrease of the fast neutrons is more rapid than that of the thermal neutrons, so the ratio of fast to thermal neutrons will decrease rapidly with distance in the reflector. The unmoderated fast neutrons (those that have suffered no collisions) will decrease with distance, x, in the moderator approximately as $e^{-x/\lambda}$ where λ is the scattering mean free path for the fast neutrons. If we take a value of 2 barns for the scattering cross section in graphite for fast neutrons, and a density of 1.6, we obtain a mean free path of 6.2 cm, which is much shorter than the diffusion length in graphite, which is 52 cm (with $\sigma_a = 4.5$ mb). The slightly moderated, but still fast, neutrons will decrease more slowly than the unmoderated neutrons, according to age theory, but still much more rapidly than the diffusing thermals in the reflector.

As already mentioned, the fast fission neutrons found inside the pile lattice or in the reflector are not very useful for cross-section measurements because of the presence of resonance neutrons, which in general have much larger cross sections than the fast neutrons. When fission neutrons are used for cross-section measurements they are usually created outside the pile to avoid contamination with resonance neutrons. A neutron "converter," consisting of a plate of U^{235} metal placed in a thermal flux, is usually used for this purpose.

Resonance Neutrons

The neutrons of intermediate energy, those in the dE/E slowing-down spectrum, are of great value for study of cross-section resonances, and are usually referred to as "resonance neutrons." Since the distance covered by a neutron in slowing down to the resonance region is larger than the distance between uranium rods in the pile, the resonance neutrons show less small-scale variation with position than either the fast or thermal neutrons, both of which have a marked variation of the order of the lattice spacing. They will thus exhibit a variation within the pile lattice close to the cosine or Bessel function predicted by the pile equation.

The number of neutrons in the resonance spectrum relative to the number of thermal neutrons is conveniently expressed in terms of the lifetime of the thermal neutrons. The number of neutrons becoming thermal per second per cubic centimeter is given by q_t, the slowing-down density just above thermal. The thermal flux resulting from these neutrons is given by $q\Lambda_{\text{thermal}}$, or by qvt_0, where t_0 is the lifetime of a thermal neutron. The relationship of the resonance flux to the slowing down density follows from moderation theory (*1*) with q_0 the number of fast neutrons per cubic centimeter that start the slowing down process, of which $q_t = q_0 p$ reach thermal. We thus obtain

$$\text{resonance flux} = (nv)_E\, dE = \frac{q}{\xi N \sigma_s}\frac{dE}{E} \quad (2)$$

This shows that the ratio of slowing-down density to resonance flux is determined by $\xi N\sigma_s$, or the slowing-down power. Calculation of the slowing-down power shows that the ratio of resonance flux to q_0, hence to fast flux, is about the same for a graphite as for a deuterium pile, even though ξ is much higher for deuterium than for graphite, because the slowing-down power is about the same for the two moderators. Because of the adjustment of the thermal absorption to attain a high k_∞, the ratio of the resonance to the thermal flux, determined by the thermal lifetime, is also about the same for a graphite- and a deuterium-moderated pile.

A convenient experimental method for the determination of the ratio of the thermal to the resonance flux is based on the cadmium ratio. The cadmium ratio is defined as the ratio of the neutron intensity, as recorded on some detector, to the intensity when the detector is surrounded by cadmium (usually about 0.03 in. thick). The bare detector responds to the resonance plus the thermal flux; when surrounded with cadmium it responds only to the resonance flux because the cadmium cover stops all the thermal neutrons but is transparent to those neutrons of energy greater than about 0.4 ev. The ratio of the thermal flux to the resonance flux in terms of the effect on the particular detector is then given by

$$\frac{\text{Thermal flux}}{\text{Resonance flux}} = R_{Cd} - 1 \quad (3)$$

where R_{Cd} is the measured cadmium ratio. The actual cadmium ratio observed is a function of the sensitivity of the detector to resonance and thermal neutrons, and does not lead to the true flux ratio unless this sensitivity is known.

The most convenient detector with a known energy variation of sensitivity is a "$1/v$ detector," in which the sensitivity is inversely proportional to neutron velocity. The specific response of a $1/v$ detector will actually be proportional to the neutron density, n, rather than to nv, regardless of neutron velocity. A foil of a material having negligible resonance absorption will have a $1/v$ cross section, and hence can be used as a $1/v$ detector if it is sufficiently thin so that it does not modify the neutron flux during irradiation. Thin foils of light elements, which have widely spaced resonances, such as B, Na, Al, would be $1/v$ detectors. Another type of $1/v$ detector is a boron-filled counter or ionization chamber, containing an amount of

boron small enough so that no flux distortion takes place.

The cadmium ratio, if measured with a $1/v$ detector within the lattice of a graphite or deuterium pile, is about 33; in other words, the resonance flux as detected by a $1/v$ instrument constitutes about 3% of the total flux. The relationship between the true flux ratio and the observed cadmium ratio of 33 can be easily evaluated from the equations of moderation theory (*1*). In general, the detector, with thermal cross section σ_{th}, will have a response to the thermal-neutron flux, $(nv)_{th}$, proportional to $(nv)_{th}\sigma_{th}$, and a response to the resonance flux given by

$$\text{resonance response} = \int_{0.4\text{ev}}^{\infty} \frac{q\sigma_a}{\xi N\sigma_s}\frac{dE}{E} = \frac{q}{\xi N\sigma_s}\int_{0.4\text{ev}}^{\infty}\frac{\sigma_a dE}{E} \qquad (4)$$

In writing the last member of this equation it is assumed than σ_s is constant; on the other hand, the absorption cross section in the "resonance-absorption integral," even without resonances, is not constant. From Eq. 3 we then obtain the general result

$$R_{Cd} - 1 = \frac{\text{Thermal response}}{\text{Resonance response}} = \frac{(nv)_{th}\,\sigma_{th}}{(q/\xi N\sigma_s)\int_{0.4}^{\infty}\sigma_a dE/E} \qquad (5)$$

which we will now evaluate for the specific case of a $1/v$ detector.

The ratio of σ_{th}, the cross section at 0.025 ev, to $\int \sigma_a dE/E$ can be easily calculated for a $1/v$ detector

$$\frac{\sigma_{th}}{\int_{0.4\text{ev}}^{\infty}\sigma_a dE/E} = \frac{1/(0.025)^{1/2}}{\int_{0.4\text{ev}}^{\infty}E^{-3/2}dE} = \tfrac{1}{2}\left(\frac{0.4}{0.025}\right)^{1/2} = 2.0 \qquad (6)$$

This result, when substituted in Eq. 5, together with the observed cadmium ratio of 33 for a $1/v$ detector, shows that the ratio of thermal flux to $q/\xi N\sigma_s$ is $32\frac{1}{2}$ or 16. The quantity $q/\xi N\sigma_s$ is, from Eq. 2, the flux in any $\log_e$ energy interval, such as 1–2.72 ev, 100–272 ev, etc., hence the flux in each such region is $\frac{1}{16}$ of the thermal flux. The flux in any energy decade, as 1–10 ev, 100–1,000 ev, etc., is, by Eq. 2, just $(\ln 10)/16$ or $\frac{1}{7}$ of the thermal flux. We can check the experimental flux ratio of 16 by an approximate calculation if we assume a thermal lifetime of 1.4×10^{-3} sec, a velocity of 2,200 m/s, and a value of 0.058 for $\xi N\sigma_s$

$$\frac{\text{Thermal flux}}{\text{Resonance flux per log}_e\text{ interval}} = \frac{qvl_0}{q/\xi N\sigma} = \frac{2.2\times10^5\times1.4\times10^{-3}}{1/0.058} = 17.8$$

The agreement is very good, considering the approximations in the theory underlying the calculated flux ratio.

Because a $1/v$ detector gives such a low response when covered with cadmium compared to the uncovered case, it is more convenient to use a material possessing a prominent resonance, hence a cadmium ratio nearer unity, even though the interpretation in terms of flux is more indirect. The usual material used for cadmium-ratio measurements is indium because of the prominent resonance at 1.44 ev. The resonance integral for indium is 2,800 barns or 14.7 times the thermal cross section as compared to a $1/v$ detector, for which the resonance integral is half the thermal cross section. Because the peak value of the indium resonance is 30,000 barns, it is necessary that the foil be extremely thin if it is desired to utilize the true value of the resonance integral. This precaution is necessary so that no self-protection (modification of the dE/E flux by the foil) will occur. As it is difficult to work with foils that are sufficiently thin, thicker foils, which must be calibrated, are usually used. The various experimental techniques that are involved in work with resonance absorption are discussed in Chapter 5 of the book.

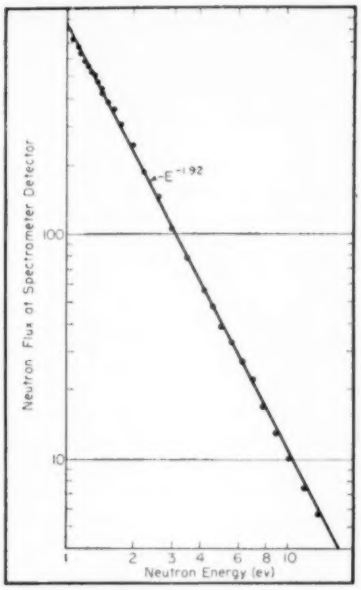

FIG. 3. Energy distribution of pile neutrons observed with a crystal spectrometer

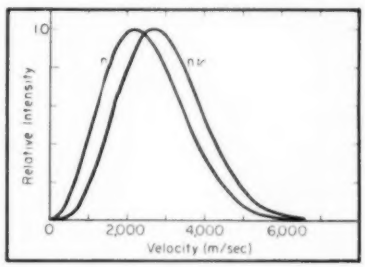

FIG. 4. Maxwell distribution of thermal neutrons

Resonance neutrons are usually used outside the pile, and to obtain high flux values, at experimental holes that are opened far into the pile lattice. The flux in such a beam, shown in Fig. 1, is calculated from the intensity at the surface from which the neutrons emerge, and from the geometry of the hole. For example, if a hole 10×10 cm is opened into the center of the Brookhaven pile where the thermal flux is 5×10^{12}, the resonance flux at a point outside the pile shield can be easily estimated. Assuming a cadmium ratio of 33, the resonance flux per energy decade in the center of the pile will be

$$\frac{5\times10^{12}}{7.0} = 7.1\times10^{11}\ \text{cm}^{-2}\,\text{sec}^{-1}$$

The flux per energy decade in the beam at a point outside the pile, say 6 meters from the inner end of the hole, will be

$$\frac{7.1\times10^{11}\times10^2}{4\pi\times36\times10^4} = 1.57\times10^7\ \text{cm}^{-2}\,\text{sec}^{-1}$$

or, as a distribution

$$\text{resonance flux} = \frac{1.57\times10^7}{2.30} = 6.9\times10^7\,dE/E\ \text{cm}^{-2}\,\text{sec}^{-1}$$

In an actual experiment, the counting rate obtained with a particular detector as a function of neutron energy, that is, the neutron spectrum as seen by the particular instrument, is of more significance than the actual flux distribution. As an example, the neutron spectrum as measured with the Brookhaven crystal spectrometer (*5*), is shown in Fig. 3. The neutron flux at the detector (corrected for detector efficiency) per unit energy interval is plotted as a function of neutron energy. As the dE/E spectrum is modified by

the reflectivity of the crystal, the distribution shown departs markedly from the dE/E flux distribution, varying as $E^{-1.92}$ instead of E^{-1}. An approximate correction of the observed curve for the crystal reflectivity can be made, however, for the reflectivity of a crystal is known to vary as E^{-1}. The result of the correction, $E^{-0.92}$, shows that the actual flux distribution checks well with the dE/E behavior predicted by theory.

Thermal Neutrons

When the neutrons are in thermal equilibrium with the moderator atoms they will have the well-known Maxwell distribution (*6*) of velocity

$$dn = \frac{4n}{v_0^3\sqrt{\pi}} v^2 e^{-\frac{v^2}{v_0^2}} dv \qquad (7)$$

where dn is the number of neutrons per cubic centimeter in the velocity range dv at the velocity v. The velocity v_0 is the most probable velocity, that is, the velocity for which $n(v)$ is a maximum as shown in Fig. 4. The value of v_0 is determined from the fact that it corresponds to a kinetic energy of kT where k is Boltzmann's constant (1.380×10^{-16} ergs per degree) and T is the temperature of the distribution

$$\tfrac{1}{2}mv_0^2 = kT \qquad (8)$$

$$v_0 = (2kT/m)^{1/2} = \left(\frac{2 \times 1.380 \times 10^{-16}T}{1.675 \times 10^{-24}}\right)^{1/2} = (1.648 \times 10^8 T)^{1/2}$$

Thus the most probable velocity at room temperature, 20° C, will be 2,200 m/s (actually 2,198 m/s) and this velocity corresponds to an energy of 0.0252 ev.

The neutrons in a distribution of temperature T are sometimes referred to as "kT neutrons" although the average neutron kinetic energy is $\frac{3}{2}kT$ ($\frac{1}{2}kT$ per degree of freedom) while kT is the energy corresponding to the most probable velocity. Thermal cross sections are usually tabulated as cross sections for a velocity of 2,200 m/s, although at times they are given for an energy of 0.0250 ev, a difference well within the error of most measured cross sections. The average velocity of the Maxwell distribution is greater than the most probable velocity, and integration of Eq. 7 shows that the average velocity is greater than the most probable by the factor $2/\sqrt{\pi}$ or 1.128. The root-mean-square velocity corresponds to the average energy, and is obviously equal to $\sqrt{\frac{3}{2}}\,v_0$.

The Maxwell distribution gives the density of neutrons, n, as a function of velocity, rather than the flux. The flux distribution, obtained by multiplication by the velocity, is also shown in Fig. 4, and is

$$d(nv) = \frac{4n}{v_0^3\sqrt{\pi}} v^3 e^{-\frac{v^2}{v_0^2}} dv \qquad (9)$$

The most probable flux, evaluated by differentiation of Eq. 9, is found at a velocity of $\sqrt{\frac{3}{2}}\,v_0$, which corresponds to a velocity greater than the most probable velocity in the Maxwell velocity distribution by a factor 1.22.

If a neutron beam is formed by a hole that opens into the reflector of the pile or into the thermal column, the flux distribution in the beam will be given by Eq. 9 because the beam flux is proportional to the flux at the neutron-radiating surface. The effective flux distribution, that is, the flux distribution as seen by a particular detector, is obtained from Eq. 9 by multiplication by the sensitivity of the particular detector. If the detector has a $1/v$ sensitivity, such as a thin boron counter, then the distribution of counting rate versus neutron velocity will again be just the Maxwell velocity distribution of Eq. 7.

The actual velocity distribution of the thermal neutrons in the pile agrees well with the Maxwellian; those differences that do exist, as well as the question of the appropriate value of T, are covered in later sections of the book.

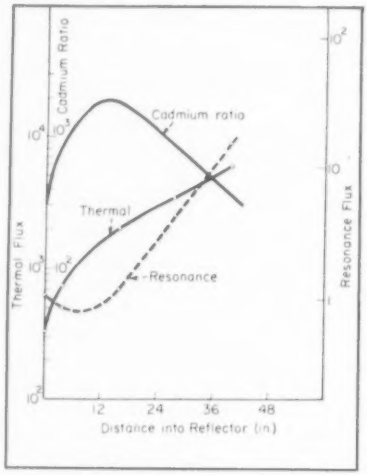

FIG. 5. Thermal and resonance fluxes observed with 1/v detectors in reflector of Brookhaven reactor

In most experiments involving thermal neutrons it is necessary to keep the contamination of resonance neutrons as low as possible. Although the highest thermal flux is found at the center of the lattice, the high resonance flux at that point means that some other location must usually be used as a source of thermal neutrons.

A compromise between high intensity of thermal neutrons and low resonance contamination must usually be made because decrease of resonance contamination is almost always associated with lower thermal flux. In the Brookhaven pile (Fig. 1), for example, the compromise is made by using the reflector as a source of thermal neutrons for all slow-neutron experiments with the exception of a few that are insensitive to resonance neutrons.

Figure 5 shows the behavior of the resonance- and thermal-neutron flux in the reflector of the Brookhaven pile as measured (*7*) with a $1/v$ detector. The detector was a BF_3 proportional counter that was sufficiently thin so that no significant self-protection was present. The curves show that the resonance neutrons decrease rapidly with distance into the reflector from the lattice, and the thermals much more slowly, the cadmium ratio thus improving with distance from the lattice. The cadmium ratio is expected to increase monotonically through the reflector but the curves show an unexpected decrease in the cadmium ratio as the outer edge of the reflector is approached.

Examination of the resonance and thermal curves reveals that the reversal of slope of the cadmium ratio results from an increase in the resonance flux at the edge of the reflector. This unexpected behavior of the resonance flux is caused by fast neutrons that leak out of the center of the pile through air-cooling channels, stream along the cooling gap, and finally migrate into the outer region of the reflector. The highest cadmium ratio is obtained for a hole that opens about one foot into the reflector, and this hole depth is used for slow-neutron choppers and mirrors, for which a low resonance contamination is essential. The cadmium ratio at the best position is 3,000, which is about 100 times that for the center of the pile, and the thermal flux at that point is about 4×10^{11} or $\frac{1}{10}$ of the central flux.

Cadmium ratios higher than those

attainable in the reflector are found in the thermal column, which is merely an extension of the reflector, usually about 5 ft square, through the shield. As the neutrons move through the thermal column, the thermal flux decreases exponentially with a relaxation length (distance for a $1/e$ decrease) slightly smaller than the diffusion length. For an infinite plane source the relaxation length is equal to the diffusion length, but leakage from the sides of the thermal column decreases the relaxation length somewhat.*

The resonance neutrons will decrease much faster than the thermals with distance in the column, and the cadmium ratio will become extremely large as the neutron intensity decreases. Because of the low intensity in the thermal column, neutrons near the surface (away from the pile) of the column are used only when an extremely high cadmium ratio is needed. For most experiments, an opening is made far into the thermal column to get the highest thermal flux consistent with the allowable resonance-neutron contamination.

The flux values as measured (*8*) with an indium foil for the thermal column of the Argonne heavy-water pile are shown in Fig. 6. The cadmium ratio shown has been corrected for the indium resonance absorption to give the cadmium ratio for a $1/v$ detector. The resonance flux, given by the activation of the Cd-covered foil, decreases approximately exponentially with a relaxation length of 13 cm.

The penetrating fast neutrons, which produce the resonance neutrons, have a scattering mean free path in graphite of about 8 cm (for a σ_s of 1.5 b, the value at 2-Mev neutron energy) and the resonance flux far from the neutron source would be expected to drop off about as rapidly as the fast neutrons, hence the observed 13-cm relaxation length is reasonable.

The thermal flux also decreases exponentially and its observed relaxation length, 30 cm, is less, because of neutron leakage out the sides of the finite thermal column, than the 50 cm (that is, the diffusion length) that would obtain for an infinite column. The actual value of the thermal flux is about 10^{11} at the inner face of the thermal column, the zero of the ordinate

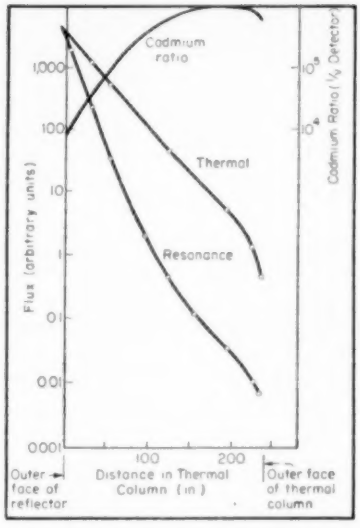

FIG. 6. Distribution of thermal and resonance flux in the thermal column of the Argonne heavy-water reactor

scale in Fig. 6, and proportionately lower at points further from the pile lattice.

The thermal flux in a beam is calculated from the geometry of the experimental hole in the same manner as the example given for the resonance flux. Thus a 10 × 10-cm hole opened to the center of the Brookhaven pile will produce a thermal flux in a beam just outside the pile shield, about 6 meters from the center of the pile, given by

$$n\nu = \frac{5 \times 10^{12} \times 10^2}{4\pi(600)^2} = 1.1 \times 10^8 \text{ cm}^{-2} \text{ sec}^{-1}$$

For the case of a beam from the thermal column, the flux at the radiating surface will be much less than that at the pile center, and this decrease will be only partially compensated by the fact that the radiating surface is closer to the point just outside the shield where the neutrons are used. For example, if the 10 × 10-cm hole is opened 2 meters into the thermal column of the Argonne pile, at which point the flux is 10^{10}, the flux at the outer end of the hole would be

$$\frac{10^{10} \times 10^2}{4\pi(200)^2} = 2.0 \times 10^6 \text{ cm}^{-2} \text{ sec}^{-1}$$

The cadmium ratio for the beam (50,000) is much higher than that in the pile lattice (33) or the reflector (3,000). For certain experiments in which a high cadmium ratio is essential, this thermal-column beam would be used even though the flux is lower by a factor of 50 than that of a beam from the pile lattice. Flux and cadmium ratios for other positions in the reflector and the thermal column can be estimated from the curves of Figs. 5 and 6.

* The actual distribution is calculated in later sections of the book in connection with diffusion measurements.

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CRITERIA OF TRACER METHODOLOGY

In successful tracer work, these seven criteria are the governing considerations: there can be no compromise among them:

Radiochemical purity	Avoid isotope effect
Single chemical state	Avoid chemical effects
Eliminate exchange error	Avoid radiation effects

Know the degree to which the tagged molecule remains intact

Yet in each experiment, three additional factors must be considered:

Economy of time and effort
Economy of radioactive material
Desired degree of accuracy

Among these factors, compromise is essential.

—AEC film, "The Radioisotope—Methodology"

A New Pulse-Analyzer Design

A versatile basic design for pulse-height analyzers having narrow channel width, stability, and high speed has been developed. The channel circuits use 6BN6 tubes. A single- and a 10-channel analyzer based on the design are described

By C. W. JOHNSTONE
Los Alamos Scientific Laboratory, Los Alamos, New Mexico

A NUMBER of new single- and multichannel analyzers of original design are being used at Los Alamos to meet the increasing need for pulse-height analyzers in the field of nuclear physics. Varying in number of channels and speed, they all stem from a basic design which has proved to be versatile and easy to build and maintain. Since 1950, more than 12 new multichannel analyzers, and even more single-channel units, have been built.

Some noteworthy features of the design are:

1. The input pulses pass through a special amplifier section which serves a purpose similar to the Oak Ridge "window" amplifier (*1*).

2. Also in common with the Oak Ridge circuits, 6BN6 tubes with relatively large bias differences are used in the channel circuits for channel-width stability.

3. The channel trigger circuits consist of one-tube univibrators which are triggered through the 6BN6's.

Although there are similarities between the Oak Ridge and Los Alamos circuits, there are also basic differences which result in considerable economy in the number of tubes in the circuits to be described.

A description of some present-day applications of pulse-height analyzers, and an independent discussion of design considerations has been given by Kelley (*2*).

Since the principles of operation of the single-channel and multichannel circuits are quite similar, the following discussion of the window amplifier and channel circuits applies, in the main, to both.

Window Amplifier

A block diagram of the single-channel analyzer is shown in Fig. 1.

Trigger circuit instabilities due to mechanical shock, temperature changes, aging, and variation in heater voltage are considered to be of the order of ± 0.1 volt. However, a channel requires two independent circuits for the upper and lower bounds and, since they may change in opposite directions, more than ± 0.1 volt may be expected. For $\pm 2\%$ channel-width stability, the bias difference between 6BN6's should probably be about 8–10 volts.

Since most applications require channels which are considerably narrower than 8 volts, a window amplifier is used (Fig. 1). Thus, 1-volt channels are achieved with a window amplifier gain of 8–10, and wider channels may be obtained either by reducing the gain or by increasing the bias difference. The former means for changing channel width is used in the multichannel circuits, but the latter is more convenient in the case of the single channel analyzer.

The behavior of the window amplifier must meet some severe requirements. It must be stable, linear for the extent of the window, deliver a large output pulse, and operate in a biased-below-cutoff condition. Furthermore, it must not be adversely affected by input pulses which may be many times greater in amplitude than the top of the window.

Another consideration has to do with large bias-to-window ratios, i.e., 1-volt channels set in the vicinity of 100 volts. In this case, only a few percent of the top of an input pulse is amplified, resulting in a rather narrow output pulse (Fig. 2). Certainly some restrictions must be placed on input pulse shape.

Pulses *A* and *B* in Fig. 2 are two of the various shapes obtainable from the Los Alamos model-500 series of mercury-relay pulse generators used for testing and calibrating pulse-height analyzers. The tops of the two pulses shown do not differ very much in width, although *B* is acceptable for analyzer calibration whereas *A* is not. For different bias settings, covering an input pulse range of 10–100 volts, if no channel changes width by more than 2%, the calibration is considered acceptable.

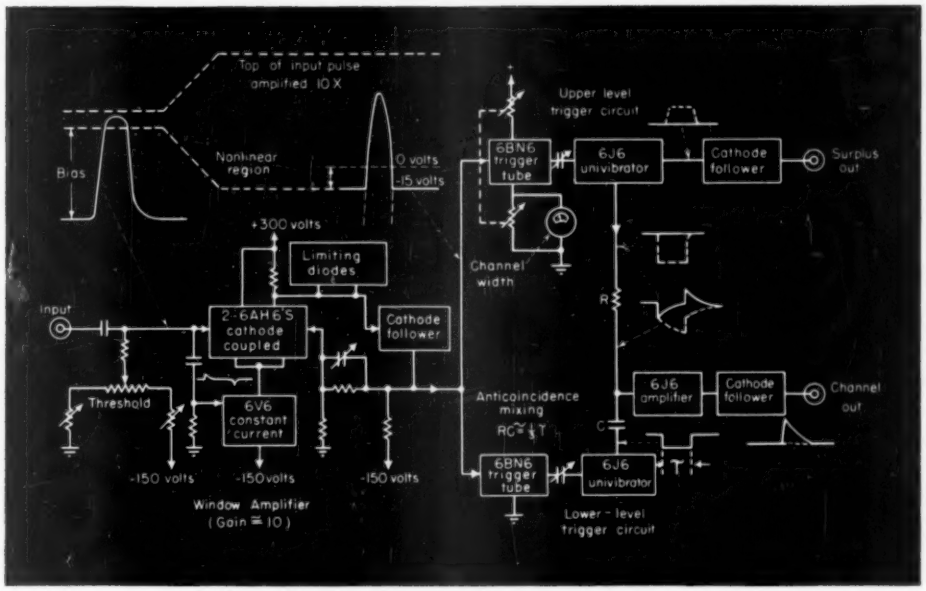

FIG. 1. Block diagram of single-channel analyzer

With pulse *A*, channel-width variations may be as much as 5 or 10% in the worst cases because of limitations in speed of response of the window amplifier as well as poorer tracking of the channel circuits. Probably the flatter the top of a pulse, the faster it may rise and yet be acceptable, but it is felt that pulse *B* is close to the practical limit on speed of rise. The 10–90% rise time of pulse *B* is 0.36 μsec compared to the window-amplifier rise time of 0.1–0.15 μsec, but one must remember that it is the top of the pulse that is important. Approximately the top 2 volts of input pulse, amplified, is required to overcome the bias of the lowest-level trigger circuit, and it appears that the rise time of this top 2 volts should not be shorter than the rise time of the window amplifier.

The input-pulse shape requirements are generally fulfilled by ionization-chamber pulses and sodium-iodide scintillation pulses. Faster rising pulses, from some proportional counters for instance, may require rise-time "spoiling." The necessary minimum pulse width need not limit the resolving time of the analyzers, for if delay-line-shaped pulses are used, (see pulse *B'* in Fig. 2) the over-all pulse width may be not much more than 1 μsec. At the other extreme, consistent operation with no recalibration has been obtained with input-pulse rise times as long as 10 μsec, making a useful range of about a factor of 20 in permissible rise times.

Circuit-wise, the window amplifier consists of a pair of cathode-coupled tetrodes or pentodes, with negative feedback applied to the grid of the

Advantages of the New Design

1. Positive input pulses are accepted over amplitude range of 3–103 volts plus sum of channel widths
2. Channel widths as narrow as 1 volt are stable to ±2%; channel position is stable to about ±0.2 volt over period of several days
3. Width and position of channels may be quickly changed; no recalibration is necessary
4. No pulse-stretching circuits are required
5. Has negligible sensitivity to pulse shape over wide range of rise times
6. Resolving time is about 4 μsec, and can be less
7. Gated operation is easily provided
8. Design is simple and economical. A 10-channel analyzer with gating needs 37 tubes plus scalers and power supply
9. No highly specialized or unusually expensive components are needed

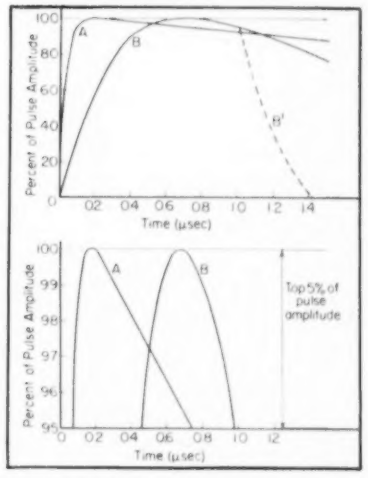

FIG. 2. Test-pulse shapes

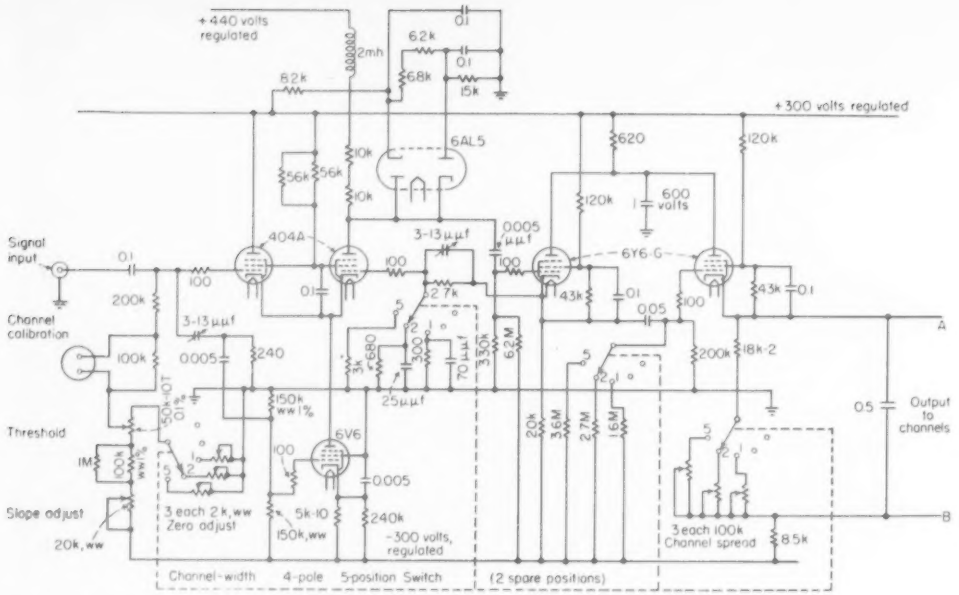

FIG. 3. Window amplifier for 10-channel analyzer

second by means of a cathode follower. Figure 3 shows the window-amplifier circuit for the 10-channel analyzer.

For this application, sharp-cutoff, high-transconductance tubes are recommended. For one or six channels, 6AH6's are adequate. For 10 or 18 channels, one may use WE 404A's and WE 418A's, respectively. A 6V6 used as a constant-current source in the common-cathode circuit helps prevent grid current during large input pulses; also the grid-cathode capacity feedthrough in the input tube is canceled with the constant-current tube by supplying the grid of the latter with a properly differentiated version of the input pulses.

In the cathode-coupled pair, although negative feedback is applied only to the second tube, the first tube also is benefited because the feedback considerably increases the cathode input impedance of the second, thus permitting the first tube to have a gain of nearly unity. By switching the feedback ratio, with suitable changes in the high-frequency compensation, the amplifier gain is changed to provide different preset channel widths for multichannel analyzers.

At low gain, the over-all linearity is quite good, but at a gain of 8 or 10 it is well to assume that the first 15 volts of pulse is nonlinear. To avoid the nonlinear region of the amplifier, sufficient bias is used on the lowest-level trigger circuit. In the multichannel circuit this bias is switched with channel width and is made approximately proportional to the window-amplifier gain so the amount of wasted pulse is constant as viewed from the input.

The plate of the second tube is connected to two diodes to limit the size of the output pulses. One of these diodes conducts a few milliamperes in the quiescent condition. For an output pulse to appear, the input-tube pulse current must exceed this amount. It might be argued that this diode current represents wasted plate current, but it is really justified by improved threshold stability and reduction of the extent of the nonlinear region.

The cathode follower which supplies feedback may also be used as the output stage in the single-channel circuit, but in multichannel circuits a second cathode follower is used as a buffer.

Adjustment of the trimmer condenser in the feedback path is made easy if a clean, flat-topped, input test pulse is available. In the switched gain circuit this adjustment is made at the setting for widest channels (lowest gain). Fixed-padding capacitors are then chosen to give the best transient response for the narrower channel-width settings.

Trigger Circuits

First, the useful and perhaps unique properties of the 6BN6 should be understood in its application as a trigger amplifier in the single- and multichannel analyzers.

As can be seen from a curve of the positive grid current of a 6BN6, it is quite easy to drive the No. 1 grid positive, since only a small amount of grid current results. The plate current changes from zero to saturation value as the grid moves through a region of about 3 volts around zero bias. As many as 19 6BN6's are driven from a single 6Y6 cathode follower in the case of the 18-channel analyzers. The grids with the least bias are driven more than 150 volts positive with no apparent ill effects.

Perhaps not obvious is the fact that with the plate current cut off by the No. 1 grid, a sizable current flows to the second grid (accelerator). This current is 5.5–6 ma in the circuits shown, and is of some use, since a small potentiometer in the cathode circuit provides enough bias variation to act as a channel-boundary adjustment which is necessary because of tube and component differences.

The plate is so well shielded from grid No. 1 that there is essentially no feedthrough. The No. 3 grid of the 6BN6 is a convenient place to insert a

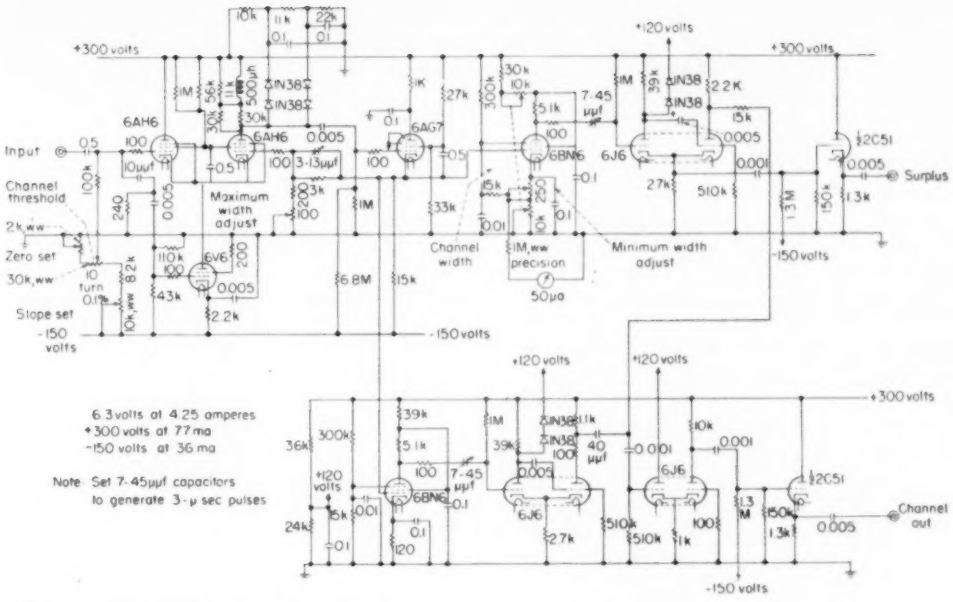

FIG. 4. Single-channel pulse-height analyzer

gate, for either coincidence or anticoincidence purposes, since it is nearly as effective as the No. 1 grid in controlling the plate current. The useful life of 6BN6's in this application appears to be about 1,500–2,000 hours.

The 6J6 univibrators are of the bootstrap variety. When triggered, more than 20 ma flows through the normally-cutoff half of the 6J6. Output pulses of either polarity may be obtained at fairly low impedance. Triggering takes place through a variable capacitor which also controls the pulse duration. If the effective width of the triggering pulse is greater than the univibrator-pulse duration, the latter will be stretched to a maximum of about 1.25 times its normal duration.

So that the trigger circuits may constitute separate channels, there must be a blanking (or anticoincidence) action upon the output of a given circuit when the next higher circuit is triggered. This action is obtained simply by mixing negative pulses of different amplitudes from the upper- and lower-level univibrators through a resistor and a capacitor, respectively (Fig. 1). The *RC* product, including stray capacitance, should be about one-third of the nominal univibrator-pulse duration, and the upper-level-pulse amplitude is chosen to be twice that of the lower. Effective canceling action is obtained even when the duration of the lower-level pulse is about 1.4 times the duration of the upper-level pulse. In the absence of a blanking pulse from the next higher circuit, a given negative univibrator pulse is simply differentiated, to obtain a delayed positive pulse, which after amplification in a biased-to-cutoff stage may be used to trigger a scaler.

The 6BN6-univibrator combination

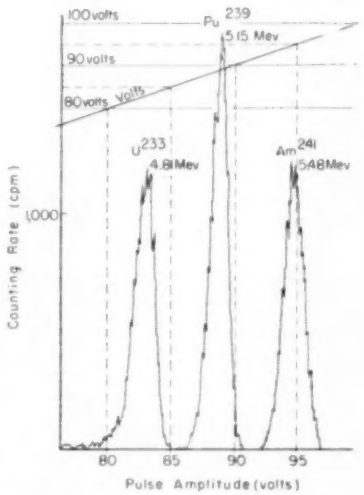

FIG. 5. Spectrum of three alpha emitters taken with a channel width of 0.8 volt

which is being used has proved to be very useful, although there are obvious limitations to this kind of a circuit. For instance, since only a small capacitor connects the 6BN6 to the univibrator, slowly rising input pulses conceivably can turn on the 6BN6 too slowly to cause triggering. At the other extreme, with input pulses which are too narrow, unavoidable differences in the response of individual univibrators will show up as poor tracking and cause changes in channel width.

A very useful range of input-pulse shapes lies between these two extremes, however. At the plate of the 6BN6, the amplitude of pulse which is just sufficient to trigger the univibrator is remarkably consistent for input pulses with rise times between a few tenths of a microsecond and several microseconds. Pulses which have wide enough tops for the window amplifier also provide adequate triggering action.

In each analyzer, all univibrators are set for equal pulse durations, and these should be at least as great as the overall rise time of the input pulses to be analyzed. If the best resolving time is sought, it follows that it may sometimes be advisable to alter the univibrator pulse durations and the anticoincidence *RC* time constants to suit different shapes of input pulses.

The multichannel circuits should

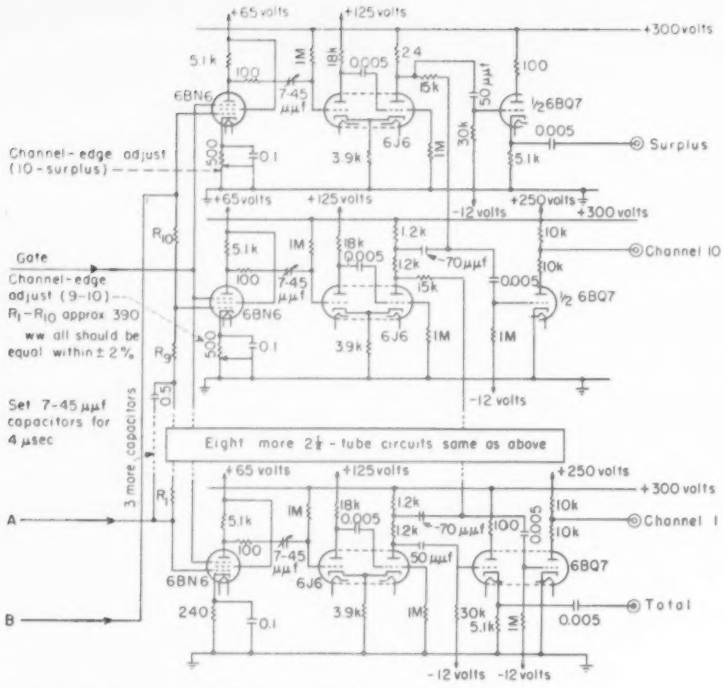

FIG. 6. Channels of the 10-channel analyzer

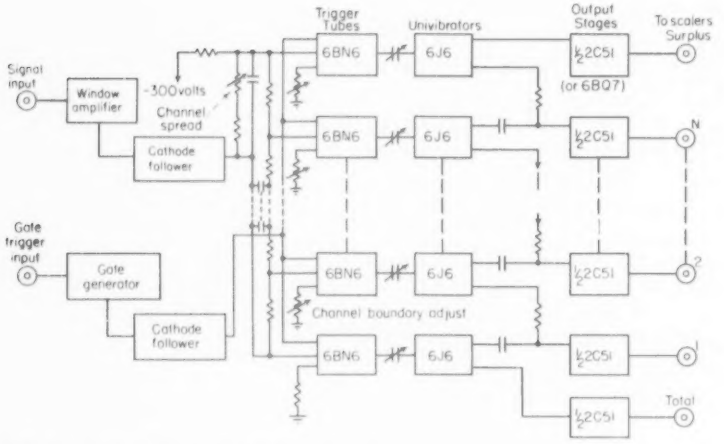

FIG. 7. Block diagram of multichannel analyzer

probably not be used to generate pulses shorter than 3 μsec, since below this value the 6BN6's cannot easily trigger the univibrators. However, at the expense of extra current in the 6J6 univibrators, obtained by using smaller plate and cathode resistors, satisfactory operation can be obtained with pulses narrower than 3 μsec, if desired.

A tolerance of about ±10% is permissible in the duration of the univibrator pulses. This tolerance could be considerably greater if it were not for the stretching effect of wide input pulses, but it has proved to be quite adequate nevertheless.

Single-Channel Analyzer

The first application at Los Alamos of the window-amplifier technique for obtaining stable, narrow channels was in a single-channel analyzer built in 1950. Our original gain-of-10 window amplifier used two cathode-coupled 12AT7 stages with negative feedback and, instead of 6BN6's, biased germanium diodes were used as pickoff elements. In the course of time, several other window-amplifier circuits were tried before the present circuit was developed. The biased crystals were soon abandoned in favor of 6BN6's to avoid temperature difficulties. Some crystal diodes remain in other parts of the circuit but cause no trouble if the operating temperature is less than 50° C. Figure 4 is the schematic diagram of the present circuit.

The position of the channel threshold is adjustable by a 10-turn potentiometer which sets the bias for the window amplifier. By properly setting the zero and slope adjustments, the 10-turn dial reads directly between 3 and 103 volts. Channel width is shown on a meter which indicates up to 5 volts.

The gain of the amplifier (nominally a factor of 10) may be varied by a potentiometer which acts as the adjustment for maximum channel width. The minimum-channel-width adjustment potentiometer in the cathode of the upper-level 6BN6 compensates for individual tube and component differences. A dual potentiometer, which varies the cathode voltage on the upper-level 6BN6 while maintaining the same anode-to-cathode voltage, is used for controlling channel width.

The channel output pulse is amplified in a cathode-coupled stage and is made available through a cathode follower to drive an external scaler. Pulses obtained from the upper-level circuit may be counted as surplus.

A more elaborate and versatile version of the single-channel analyzer has been built for use primarily as an alpha-energy analyzer. This equipment is used together with a recorder which plots counting rate vs pulse height. The bias potentiometer is geared to the chart-drive mechanism in the recorder, which takes 10–15 minutes for a complete scan. Once started, the device requires no attention since the bias voltage is plotted simultaneously by means of a second pen. The analyzer chassis includes a counting-rate circuit and a scale-of-100. The latter is useful when making a count on such a weak sample that the automatic-scan feature cannot be used.

The channel width may be chosen from less than 1 volt to 10 volts, and the extent of the scan may be varied. Figure 5 is an example of a record taken with a channel width of 0.8 volt. It was obtained from three separate alpha-emitting samples in a chamber with 50% counting geometry.

In at least one instance to date, the single-channel analyzer has been successfully duplicated and used in another laboratory (*3*).

Multichannel Analyzers

The single-channel circuit served as a useful stepping-stone towards the design of a variety of multichannel analyzers. These have been built with 6, 10, and 18 channels. Optional features have included the choice of scales-of-10 or scales-of-100 for each channel, motor-driven register-reset mechanism, and an input-signal delay which is often necessary for coincidence work.

It would be desirable, of course, if analyzers having 50 or 100 channels could be made available in quantity, but no units having more than 18 channels have been attempted with the present design (*4*). Actually, the 10-channel units with scales-of-100 are finding the greatest demand. A large amount of information can be obtained in a short time with a 10-channel analyzer when one has stable, high counting rates, because it is easy to slide channels from one position to the next.

Operation of the window amplifier and channel circuits has been described above; the multichannel circuit diagrams are given in Figs. 3 and 6. A block diagram is given in Fig. 7, and the gate generator which is a part of every multichannel analyzer, is shown in Fig. 8.

The count-stop switch in Fig. 8 places a large negative bias on the No. 3 grids of the 6BN6's to stop the action of the analyzer. The No. 1 grids of all the 6BN6's receive the same amplitude of pulse from the window amplifier, but the bias for each 6BN6 differs by 8 or 9 volts from its immediate neighbors. The over-all bias difference between the first and last 6BN6's is separately variable for each channel-width setting, so that there is an adjustment of about $\pm 5\%$ in the spread or sum of all the channels. Fine adjustments for each channel boundary are made with 500-ohm potentiometers between 6BN6 cathodes and ground.

The 18-channel circuit is similar to the 10-channel except for the use in the window amplifier of WE 418A's, which can develop an output pulse of about 175 volts, with a rise time of close to 0.1 μsec.

Gating

The gate generator shown in Fig. 8 resembles the channel trigger circuits, since it is made with a 6BN6 and 6J6 univibrator. Bias control on the 6BN6 is provided with a calibrated discriminator dial which reads trigger level in volts.

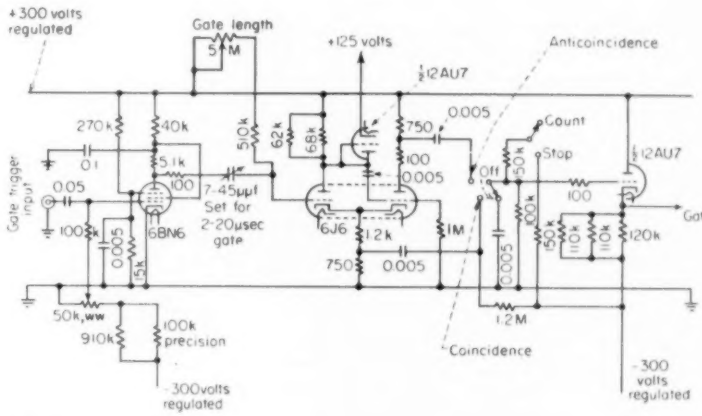

FIG. 8. Gate generator

The gate may be used in the following ways:

1. When it is set for coincidence, the channel circuits will not function unless the gate is triggered in coincidence with the analyzer input pulses. Positive gate pulses go to the No. 3 grids of the 6BN6's to overcome the negative bias which is applied in this case.

2. When it is set for anticoincidence, the operation is just the opposite of **1**, and there is a negative gate and positive bias on the No. 3 grids.

3. It may be used for self-gating. This is beneficial in cases where long-tailed input pulses must be used, and the noise level at the input is appreciable. If no gating is used, occasional false triggering of channels may occur if the effective width of an input pulse is greater than the duration of the univibrator pulses. To prevent false triggering of the channels by noise, the analyzer input pulses are used to trigger the gate which is set for coincidence. The gate width should be a little greater than the over-all rise time of the input pulses, and the gate discriminator dial should be set to a lower value than the analyzer threshold dial. When this is done, the channels are gated off soon after the peak of the input pulse has passed, and a new gate cannot be generated until the tail of the input pulse has dropped well below the threshold of the first channel. There remains the probability that some false gates may be triggered, but a small percentage of false gates is far less serious than false counts in the channels.

Self-gating is unnecessary if all input pulses, preferably delay-line shaped, are of shorter duration than the channel univibrator pulses.

* * *

The author is grateful to the many people who have supplied help and encouragement in this work. Carroll W. Zabel and Bob E. Watt provided most of the original inspiration, and both contributed many helpful ideas. O. L. Stone has helped immensely by providing precision pulse generators and scalers.

The author also wishes to thank Walter L. Jordan of Oak Ridge National Laboratory for first introducing him to the 6BN6 tube.

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Low-Temperature Steam Power Cycles for Water-Cooled Reactors

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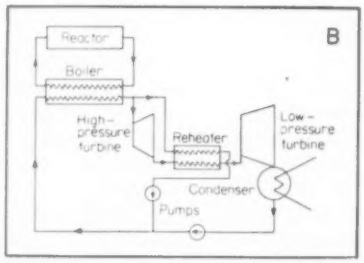
LOW-TEMPERATURE saturated steam cycles are not used in combustion-fired power plants because superheated steam cycles at higher temperatures are more efficient. But, with regenerative heating to raise efficiency, they appear to be practical for use with nuclear reactors when the reactor design seriously limits the available steam temperature. Excess moisture in the turbine is a problem, but it can be avoided without loss of efficiency. Methods for doing this and the choice of a saturated-steam cycle are considered.

2. REDUCE MOISTURE that appears when initially saturated steam expands through a turbine. Turbine efficiency decreases about 1% for each 1% of moisture. Also, moisture erodes turbine blades; Westinghouse (private communication) estimates life expectancy of stellite-protected blades to be 10 years for 12% moisture, 6 years for 15%, and possibly 3 years for 18%.

Literature values for the maximum permissible moisture in the turbine exhaust are all near 12%. But expansion of 70-psia saturated steam results in 12.6% moisture at a condenser pressure of 1½ in. Hg. At the higher pressures required for efficiency, the moisture content is even higher.

Turbine design cannot eliminate the moisture problem. At the expense of cycle efficiency, a higher condenser pressure, throttling the steam, or superheating can. Fortunately, turbine moisture can be reduced by the methods at right without loss of efficiency.

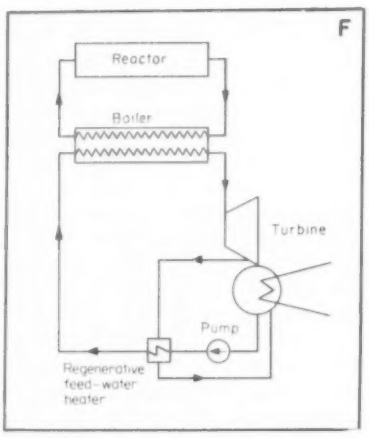


REHEATING WITH BOILER STEAM after partial expansion can reduce moisture without impairing efficiency. Actually, efficiency can be increased (see G), but turbine exhaust moisture then increases to 12%

3. INCREASE EFFICIENCY of the steam cycle by using regenerative feed-water heating. Flow diagram for one stage of regeneration is shown at right in *F*. Each additional regenerative feed-water heater adds less to the efficiency of the cycle, so an economical number of heaters exists for any set of conditions. In conventional central power stations, not more than four or five are used. The steam for each successive stage is bled from the turbine at a more advanced point.

If the temperature rise of the reactor coolant is large, regenerative feed-water heating might necessitate a reduced steam pressure to allow heat transfer from the reactor coolant at its lowest temperatures. The cycle with the optimum efficiency may be a compromise between the maximum pressure and the maximum amount of regeneration possible.

For a large central power station in which a water-cooled reactor furnishes the energy, a saturated steam cycle could be used, with a separator to reduce turbine moisture and four feed-water heaters to raise cycle efficiency. Based on calculations for a 100,000-kw reactor, with saturated steam generated at 450 psia, it is estimated that the usable electric power would be 26,450 kw, less the amount required to pump the reactor coolant. If 1,000 kw are sufficient for this, the net plant thermal efficiency will be 25.4%.



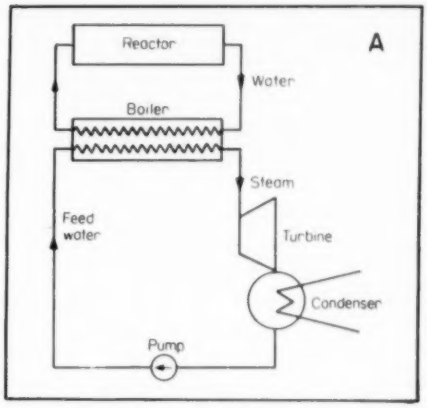
REGENERATIVE FEED-WATER HEATING is most important improvement on Rankine cycle. One stage is shown here

1. DERIVE ENERGY from a heterogeneous reactor cooled by pressurized water. The reactor coolant generates steam in a heat exchanger, and the steam is used to produce electric power. (However, discussion is also applicable to other systems.)

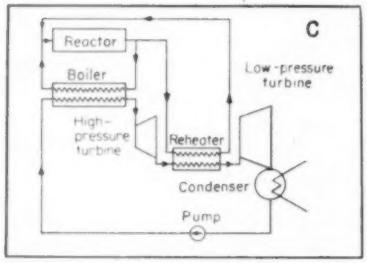
For heat transfer to occur between the reactor coolant and the steam in the boiler, a temperature difference must exist. The coolant temperature is fixed by heat transfer requirements in the reactor and by boiling prevention; it can be considered to be less than the critical temperature of 705° F. Steam temperature must be lower; for the system at right, the maximum steam temperature probably would be about 500° F.

For steam temperatures less than about 500° F and condenser pressures greater than about ½ in. Hg (59° F), the most efficient steam cycle will be one using saturated rather than superheated steam. When the maximum steam temperature is limited, superheating must be accomplished at the expense of pressure, thus adversely affecting cycle efficiency.

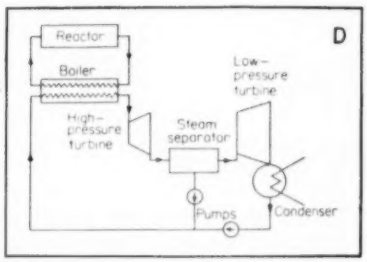
Confirmation of the superiority of the saturated steam cycle in this range has been obtained from a Mollier chart with condition lines which include moisture losses.



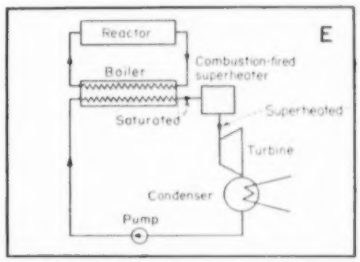
RANKINE CYCLE for deriving power from a water-cooled reactor is used with system shown. Cycle can use saturated or superheated steam. When steam is limited to a temperature less than 500° F, the saturated cycle is more efficient



REHEATING WITH REACTOR COOLANT after partial expansion is more efficient than using boiler steam. But radiation can seriously complicate piping and shielding, making use of steam preferable



MOISTURE SEPARATOR between high and low-pressure turbines appears advantageous, since considerable gain in cycle efficiency can be obtained at small cost. Separator can remove 90% of moisture

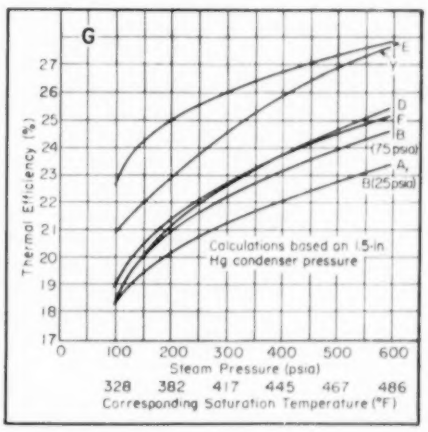


SUPERHEATING with conventional fuel can eliminate moisture and also raise efficiency. But plant will be complicated and the superheater expensive because of the low heat-transfer coefficients needed

4. COMPARE saturated and conventional steam power plants. Efficiency curves are shown in *G* for saturated steam plants with moisture control and regeneration. Curve *Y* is for four stages of regeneration plus a separator.

By comparison, a plant efficiency of 30–35% might be expected for a conventional plant at best load, 400–600 psia, 700–800° F and four stages of regeneration. About 15% of heat of combustion will be wasted in conventional plant, but loss in boiler heated by reactor coolant is negligible. Even so, efficiency of low-temperature saturated steam plant should always be lower than that of conventional plant at the same pressure. However, with steam temperatures as low as 328° F (100 psia sat.), cycle efficiencies are still attractive for power production.

R. D. Elliott (NU, Feb. '52, p. 57) has shown that lowest investment and operating costs in conventional plants, exclusive of fuel, obtain with steam temperatures of 600–800° F. At 500° F, both costs rise about 10% above the minimum; at 300° F, they rise about 50%. On this basis, a 600 psia, 486° F, saturated steam plant appears economically feasible, but a 100 psia, 328° F plant appears dubious.



EFFICIENCY CURVES are lettered to correspond to flow diagrams, with *Y* for four regenerative stages plus separator, *E* for 800° F combustion superheat

Two Liquid Scintillation Neutron Detectors

Neutrons can be detected efficiently by liquid scintillators containing borates. An efficient slow-neutron counter, now being tested, has a response time of 0.5 μsec and some γ-sensitivity. A design is suggested for a fast-neutron counter that is insensitive to γ-rays and slow neutrons and should have a response time of 5 mμsec

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AN IDEAL neutron detector would be 100% efficient, respond in a very short time, be insensitive to γ-rays, and have good energy resolution.† Liquid scintillation techniques have made possible the construction of detectors closely approaching these characteristics (*1*).

The two counter designs which will be described, one efficient for slow neutrons and the other for fast neutrons, satisfy the first three ideal characteristics but are deficient in the fourth. That is, they do not, by themselves, measure the energy. However, both are suitable for use with time-of-flight equipment.

To behave ideally, detecting materials must have high atomic density and low atomic number. They should also have strong neutron interaction, resulting in a single measurable physical event of magnitude dependent on the energy. This event, a pulse of collected ions or light, should involve many separate small events (production of ion pairs or light quanta). The latter insures good energy resolution since the resolution is proportional to the square root of the multiplicity of the smaller events.

Use of Liquid Media

Since the photomultiplier and organic scintillators used here respond in about 10^{-9} sec, it was decided to try to introduce a strong neutron absorber into a fluorescent organic medium without quenching the fluorescence. An organic compound of boron was the natural choice since its neutron-capture cross section is very high (4,016 barns at 0.0253 ev for B^{10}), and capture results in an exothermic reaction yielding two energetic charged particles

$$B^{10} + n \begin{cases} \rightarrow Li^{7*} + He^4 + 2.3\ \text{Mev} & (93\%) \\ \rightarrow Li^7 + He^4 + 2.8\ \text{Mev} & (7\%) \end{cases}$$

Furthermore, the atomic number of boron is low, and the chemical and physical properties of some of its organic compounds are favorable. It forms clear colorless liquids having a high transmission for visible and ultraviolet light.

The series of boron esters, e.g., trimethyl, ethyl, propyl, and butyl borates, when added to a 3-gm/l-terphenyl 5-mg/l-diphenylhexatriene solution in phenylcyclohexane, reduce the pulse height from either alpha particles or electrons only in proportion to the

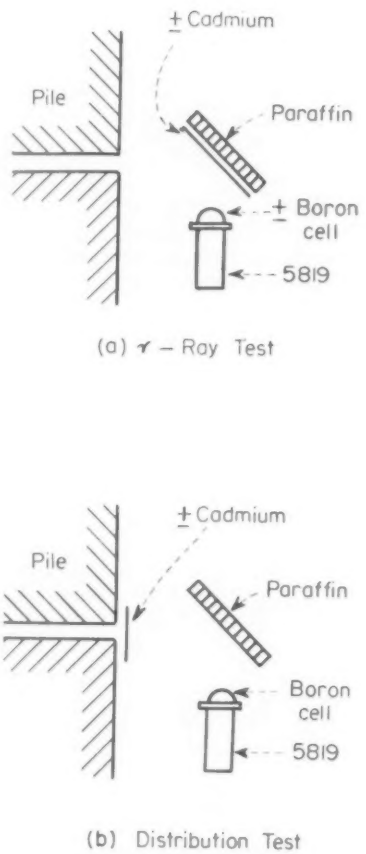

FIG. 1. Set-up for determining (a) γ-ray response and (b) slow-neutron response with pile beam

* PRESENT ADDRESS: Brookhaven National Laboratory, Upton, New York.

† The list does not include all conceivable properties, such as neutron-polarization indication.

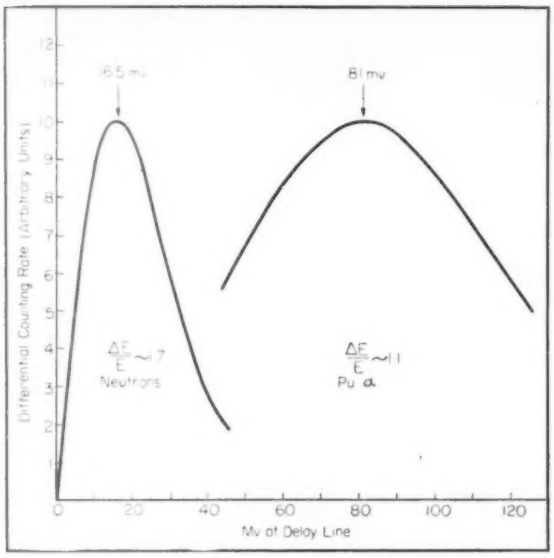

FIG. 2. Pulse-height distribution using *p*-diphenylbenzene, 900-volt potential

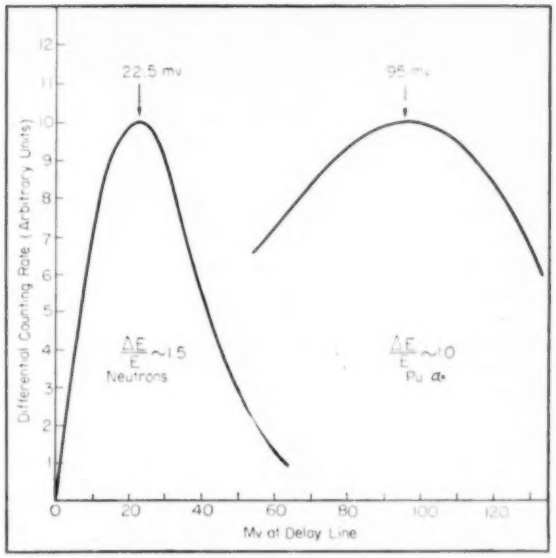

FIG. 3. Pulse-height distribution using 2,5-diphenyloxazole, 900-volt potential

amount of dilution of the original fluorescent material (*2*). The boron esters are effectively inert material in the medium.

A generalization of Kallmann's formula (*3*) for the dependence of pulse height on the concentration of fluorescent material, with the proper solvent concentration as a variable, is

$$h = NP_0C_0/(Q_0 + C_0)(R_0 + NC_0)$$

where h = pulse height; N = inert-material dilution factor, ≤ 1; P_0, Q_0, and R_0 are Kallmann's parameters for $N = 1$ (i.e., undiluted); and C_0 = undiluted concentration of fluorescent material.

This formula is derived on the assumption that the internal-quenching probability of the proper solvent is much smaller than its self-quenching probability. It is interesting that Kallmann's identification of $R_0 \gg Q_0$ is consistent with the above, since it was observed that $h \propto n$. Therefore $R_0 \gg NC_0 \sim Q_0$.

The reduced pulse height may be increased 50% by the addition of more terphenyl and diphenylhexatriene. The best terphenyl recipe to date for equal volumes of methyl borate and phenylcyclohexane and for solubility to $\sim -10°$ C is 4 gm/l *p*-diphenylbenzene (terphenyl) and 8 mg/l diphenylhexatriene. A better recipe employs 2,5-diphenyloxazole, which, for equal volumes of methyl borate and phenylcyclohexane and for solubility to $\sim -20°$ C, is 4 gm/l 2,5-diphenyloxazole and 16 mg/l diphenylhexatriene. The latter solution gives ~30% larger pulse heights than does the first one. Both solutions were prepared for use with an RCA 5819 photomultiplier.

Methyl borate is the preferred ester since, of the series, it has the greatest density and the smallest number of extraneous atoms.

Efficiency and Lifetime

To date only normal boron (18.8% B^{10}) material has been used. A counter ~2 in. in diameter and ~1 in. long was placed in the scattered thermal-neutron flux of the Argonne heavy-water reactor. Integral counting rates were taken at sufficiently low bias to detect essentially all scintillations occurring in the counter volume. A comparison of this counting rate was made with that of a thermal-neutron flux meter. Such a comparison was far from accurate but indicated an efficiency ~100%. Cadmium differences were always taken to insure that the pulses were to be attributed to a neutron effect.

The efficiency should be limited only by the nuclear processes of scattering, moderation, and escape before capture. To measure this more accurately, two cells 1¾ in. in diameter and 1¾ in. long were prepared. In one of these, equal volumes of phenylcyclohexane and ethyl alcohol were placed, and in the other, equal volumes of phenylcyclohexane and methyl borate.

The cells were placed one at a time in the center of an annular-shaped BF_3 neutron counter (*4*). A 1/16-in.-diameter neutron beam from the pile was then passed through the annulus, and the rate of escape of neutrons from each cell was observed. The cells contained so much scattering material that transmission through the cell was essentially zero. Counting rates were measured with and without cadmium in the beam.

The cell containing no boron scattered essentially all neutrons into the BF_3 counter and thereby gave a measure of the total flux. The cell containing boron captured a large fraction of the beam and allowed a smaller fraction to escape. A comparison of counting rates from the two cells permitted calculation of the efficiency. The results obtained for the fraction captured were thermal, 90%, and epicadmium (those having energies above thermal), 60%.

It is difficult to make an accurate estimate of the corresponding fractions for enriched material, i. e., an increase in capture by a factor ~5 for the same scattering. If, however, one does so on the basis of a mean escape life, τ_a, when no boron is present, the prob-

ability of capture would be $\tau_c/(\tau_s + \tau_c)$. Here τ_c, the mean life for capture by boron, is inversely proportional to the concentration of B^{10} atoms. On this basis the thermal and epi-cadmium efficiencies for enriched material are calculated to be ~98% and ~89%, respectively. These values are, of course, functions of the neutron energy and of the shape and size of the cell.

The mean life for neutron capture in the energy region in which the $1/v$ law for boron is operative is ~0.42 μsec if enriched material is used and ~2.2 μsec if normal material is used. Shorter detection times may be obtained only at the sacrifice of efficiency, e. g., by making the counter thinner. It is desirable to use enriched boron to make capture in boron as predominant as possible in the shortest possible time.

Gamma Sensitivity

Though the counter is made of materials of low atomic number, a significant response to γ-rays via the Compton effect is present. Both experiment and calculation indicate this to be ~2%/gm/cm² in the bias interval necessary for detection of boron-disintegration pulses.

An actual experiment which gave the figure 2%/gm/cm² was as follows. Two counters were compared in a scattered thermal beam from the Argonne heavy-water reactor. The first counter contained sufficient ethyl alcohol to give γ-ray pulses of the same size as those from a boron counter. The second counter consisted of the diphenyloxazole-and-diphenylhexatriene recipe given earlier.

Counting rates were taken from both counters with and without cadmium between the beam and scatterer as shown in Fig. 1a. On placing the cadmium in position, the first counter's rate increased by 50%, and the second counter's rate decreased to 20%. Assuming four capture γ-rays per neutron (*5*), the relative efficiency of γ-rays to neutrons can be calculated as follows:

The creation of four extra γ-rays with a 50% increase in gamma counting implies the presence of 12 γ-rays when cadmium is in place, and 8 γ-rays when it is not in place. These 12 γ-rays are evidently equivalent to

$$12\gamma \equiv 1\tfrac{1}{2}(n + 8\gamma)$$
$$n \equiv 52\gamma \text{ or } \gamma/n \simeq 0.02$$

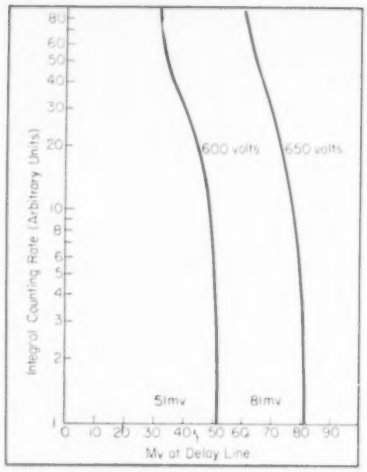

FIG. 4. Cobalt-60 γ-ray endpoint using 2,5-diphenyloxazole

Pulse Height and Noise

The setup shown in Fig. 1b was used to observe the pulse-height distribution resulting from boron disintegration. In Figs. 2 and 3 are shown the pulse-height distribution obtained and, for comparison, the pulse-height distribution from a plutonium α-particle (*6*). The latter was obtained by washing a benzene solution of plutonium into the counter. Comparisons were also made with electrons of known energy (Fig. 4). These were measured by observing the Compton endpoint of the most energetic γ-ray from Co^{60} (1,120 kev).

All measurements were made using the same aluminum cell to contain the liquids and with the same 5819 photomultiplier tube operating at various voltages. Pulses from the cathode follower were shaped by 12½ ft of RG65U and fed into an amplifier having a gain of 1,200 (Fig. 5). The multiplier gain-vs-voltage power law

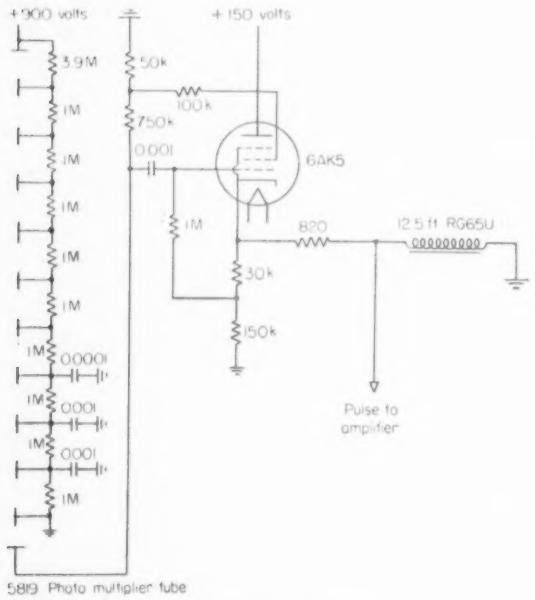

FIG. 5. Cathode-follower circuit with delay line

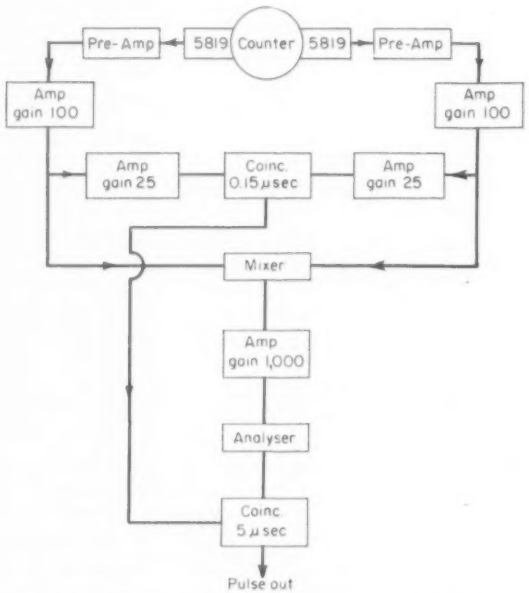

FIG. 6. Pulse-height selection for slow-neutron detection

was determined (Fig. 4), and the Co^{60} γ-ray endpoint was scaled to the 900-volt operating potential needed to observe the boron-disintegration pulses. This comparison indicated that the energy released in neutron capture was about one fortieth as effective in producing light as the energy released from fast electrons. In addition, this meant that on the average 10 to 15 cathode electrons were emitted per neutron captured.

All measurements were made with dry ice and acetone slush surrounding the 5819 tube. Under the operating conditions, this resulted in a noise count of $\sim$100 cps. Without cooling, the noise count was $\sim 10^4$ cps.

Slow-Neutron Detection

Most slow-neutron (<100 kev) detection problems do not require the counter to respond in less than 0.5 μsec. If a faster response is required, it can be obtained, but only with a proportionate sacrifice in efficiency. In any event the small pulse sizes of the scintillation liquid require three features in the counting system: gain and pulse discrimination, noise elimination, and efficient light collection.

Gain and pulse discrimination. A photomultiplier such as the 5819 is suited to slow-neutron detection since it does not require driving all pulses to saturation, i. e., to the same size. Even though pulse-height discrimination is poor, it is worthwhile to reject all pulses greater than a certain size. For example, in operating under the conditions prevailing for the distribution shown in Fig. 3, it is desirable to accept pulses only up to a bias level of $\sim$50 mv. In this way most Compton-recoil electrons are not counted. Furthermore, to make sure that only an insignificant fraction of pulses is lost at the low end of the distribution, it is necessary to effect about a 5-mv sensitivity in the amplifier and to operate the phototube at a sufficiently high voltage so that essentially all boron-disintegration pulses are counted.

Delay-line shaping may also be advantageous as it results in a smaller and more constant deadtime. The deadtime which does result should depend only on the relaxation time of the high-level discriminator.

Noise Elimination. Thermal noise from the multiplier can be eliminated either by cooling the photomultiplier or by employing two photomultipliers in coincidence. Certain problems in which a low background rate is important may require both techniques. Cooling the tube to $\sim -10°$ C is sufficient and makes much simpler demands on the resolving time of the coincidence circuit. Figure 6 shows a scheme used at Argonne National Laboratory for accomplishing this.

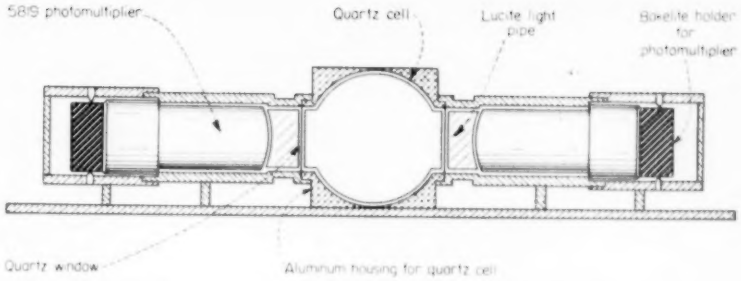

FIG. 7. Cross section of coincidence-counter arrangement

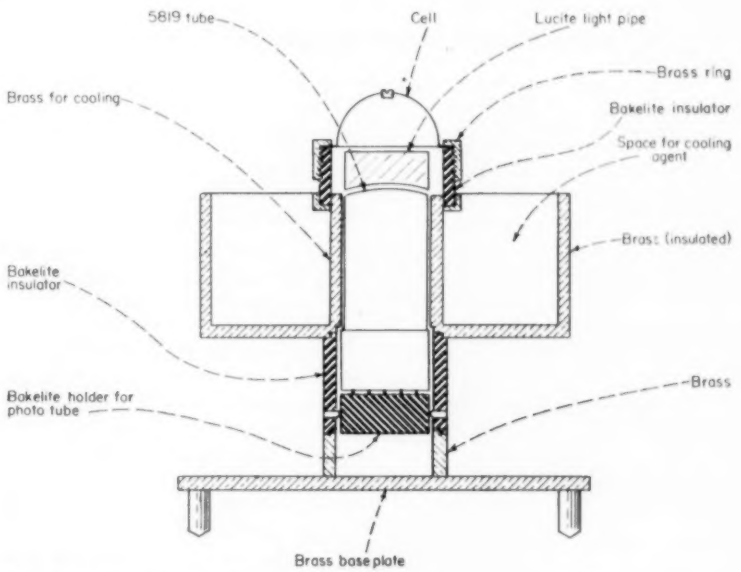

FIG. 8. Cross section of single-end-cooled counter

Optics of Light Collection. Counters of large volume have, in general, reduced pulse heights due to either internal attenuation or limited surface reflectivity. The latter effect is the more important factor, but the one most readily overcome.

To estimate the fraction, f, of light absorbed by a photomultiplier window which looks at a large counter cell whose inside surface has a reflection coefficient r, let us assume the light to be randomly scattered after each contact with the surface of the cell. Let the internal transmission factor per mean cell dimension be b, and let the ratio of the area of the window to the total area of the inside surface be a. Then $f = ab/[1 - rb(1 - ab)]$. Thus, if $b = 1$ and $a = 0.1$, and if f is 0.9, then r must be 0.99.

The problem of surface reflection may be overcome by the use of quartz cells coated on the outside with MgO. The reflection coefficient of this material is >0.97. However, polished aluminum 61S is fair for cells having mean dimensions $\sim$10 cm.

Figure 7 shows a simplified design of a quartz cell with two windows and two photomultiplier tubes in coincidence. The cell was smoked with MgO and placed in a deep-freeze operating at

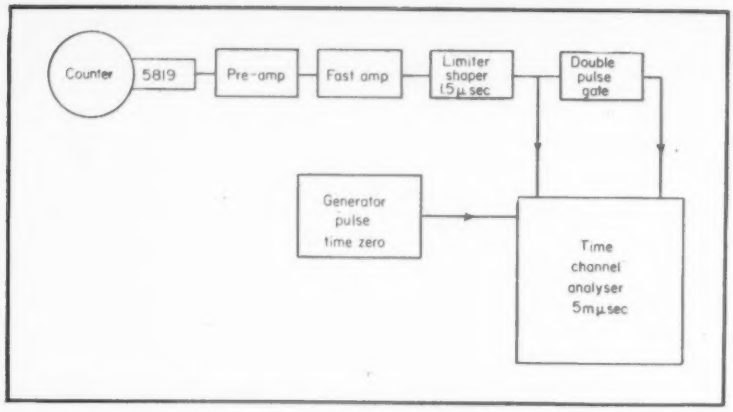

FIG. 9. Double-pulse selection for fast-neutron detection

$-10°$ C. A 5-in.-diameter hole was cut in the front face of the deep-freeze and patched with 28 aluminum. This provided entry for a neutron beam. The counting system is being tested with a resonance neutron time-of-flight apparatus functioning at the ANL heavy-water reactor.

Figure 8 is a simplified design of a single-ended system which was used at ANL in making most of the measurements described in this paper.

Boron Chemistry

Since B^{10} is available from Oak Ridge National Laboratory only as $CaF_2{\cdot}BF_3$, it is necessary to convert this into $B(OCH_3)_3$. A reaction giving a good yield is necessary because of the high cost of the material. One that shows promise is the following:

$$BF_3(C_2H_5)_2O + 3NaOCH_3 + CH_3OH \rightarrow 3NaF + [B(OCH_3)_3 + CH_3OH] + (C_2H_5)_2O$$

$$[B(OCH_3)_3 + CH_3OH] + H_2SO_4 \rightarrow (H_2SO_4 + CH_3OH) + B(OCH_3)_3$$

where the brackets indicate an azeotrope and the parentheses indicate a liquid layer immiscible with the $B(OCH_3)_3$.

Fast-Neutron Detection

The following technique has not been tried to date, but is one that the authors feel will be fruitful.

Double pulse. When fast neutrons enter a medium such as the borates, two pulses separated in time by $\sim$0.5 μsec should be detectable. The first pulse occurs in a time $\sim 5 \times 10^{-9}$ sec and is produced by a series of protons recoiling from the elastic scattering of neutrons by protons. The second pulse follows the first in a time $\sim$0.5 μsec and is a result of neutron capture by boron. Thus the counter will respond in a time $\sim 5 \times 10^{-9}$ sec (includes fluorescent lifetime), and the neutron can be identified as such by the appearance of the second pulse.

This permits discrimination against a high background of γ-rays and slow neutrons. Such a counter would be useful as a high-speed detector of fast neutrons in a time-of-flight velocity-selection apparatus.

Noise elimination. A single-ended system is satisfactory. However, cooling of the tube will again be required in order to reduce the chance rate of double pulses. A problem more serious than that of thermal noise may be the one of satellite pulsing (7). This problem can perhaps be overcome by selection of tubes.

Self-coincidence circuitry. If 5819 tubes are employed, it will be necessary to limit all pulses which originate from more than $\sim$3 cathode electrons to the same height. This may be done with a fast high-gain amplifier and limiter. If, however, photomultiplier tubes having more than 10 dynodes are used, a proper saturated pulse may be obtained directly from the multiplier.

All pulses should be rectangular in shape and have a time duration $\sim$1.5 μsec, i. e., several mean capture lives. A simple gating circuit, as in Fig. 9, with a rise time of $\lesssim$0.1 μsec should then receive the above output and permit only coincidence pulses having a height between 1 and 2 units to pass. The counter can be used in a fast time-of-flight apparatus for determining the energies of the neutrons giving rise to the first short pulses of the coincident pairs.

* * *

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Radiation vs Trichinosis

Trichinosis is a disease caused by a microscopic parasitic worm transmitted to humans through infected pork when the meat is insufficiently cooked to kill any trichina larvae that may be present. Death can occur in acute cases.

Up to now, this disease has been controlled by four methods: microscopic inspection of meat, boiling of garbage fed to hogs, low temperature treatment of pork and pork products to kill the larvae, and sufficient cooking. But the methods whose use can be insured by public health enforcement are economically impractical.

Use of radiation to sterilize the larvae could solve the problem. According to H. J. Gomberg and S. E. Gould of the U. of Michigan, X-rays, Co^{60}, or fission products could be used to irradiate whole pork carcasses at packing plants.

A High-Resolution Detector for High-Intensity Radiation

Good resolution in the detection of high-intensity radiation is hard to obtain. The detector described here, an RCA C7154 coaxial phototube surrounded by scintillator, has a resolution of 2 × 10^{-9} sec, and an output current linear with light input up to 20 amperes. Design and construction of the detector is discussed in detail

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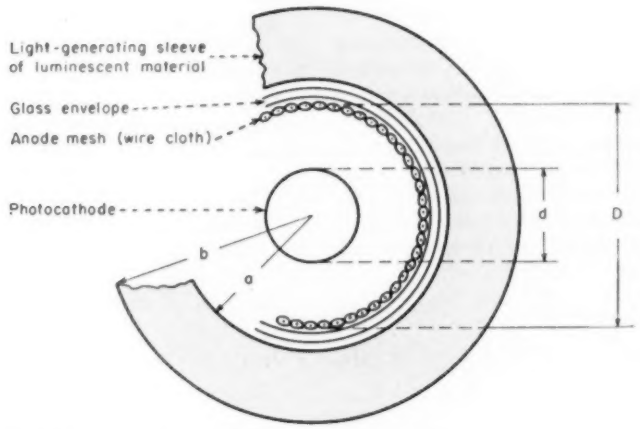

FIG. 1. Cross section of phototube and sleeve

THE REQUIREMENT of good resolution in the measurement of high-intensity nuclear radiation imposes stringent standards on both detector and circuit. Scintillation detectors can be built to achieve the high resolution needed.

A partial solution to the problem has been found by Rossi and Staub (*1*) who designed ionization chambers which can be matched into transmission lines for good high-frequency response and oscilloscope presentation. One disadvantage of this type of detector is that it functions as an integrating device; consequently, for varying signals there is considerable difficulty in analyzing the records.

The scintillation detector described here overcomes this disadvantage for signals whose rate of intensity variation is small compared to the resolution of the detector (about 2.5×10^{-9} sec). It has also been designed for large signal output so that the need for amplifiers is eliminated.

Figure 1 shows a cross section of the scintillation detector. It consists of a coaxial phototube with an opaque photocathode as center conductor and a mesh-type anode as outer conductor. Surrounding this phototube is a cylindrical sleeve of scintillating material which extends beyond the photocathode on each end to give more uniform illumination of the photocathode. Since the decay time of the organic scintillator is of the order of 2.5×10^{-9} sec, and the dimensions of the phototube are approximately 30 cm (corresponding to light paths of less than 10^{-9} sec), the decay time is the major factor in determining the resolution of the detector.

Photocathode Light Collection

The scintillation detector operates from the light produced by conversion of gamma radiation to light in the sleeve of scintillating material. If it is assumed that the distribution of light radiated from the emission centers is

isotropic, the output current of the phototube can be found in terms of the volume density of the generated light, the geometry of the sleeve and phototube, and the sensitivity of the photocathode. These results, supplemented by experiments, have been used in determining the optimum size of the phototube and scintillating sleeve. The actual sensitivities of the complete detectors were measured by exposing them to a calibrated source of gamma rays, but will not be reported here.

To simplify the computations, it is assumed that the light is generated uniformly throughout a long sleeve with a power density of ρ watts per unit volume. The light generated in the volume element shown in Fig. 2 is $dl = \rho r d\theta dr dz$. For a cylindrical sleeve of thickness dr and unit length the amount of light is

$$\int_{z=0}^{1} \int_{\theta=0}^{2\pi} dl = 2\pi\rho r dr \qquad (1)$$

It is now assumed that the generated light is perfectly diffuse; no light is reflected from the surface, and none is absorbed within the crystal sleeve. Then the fraction of that light given by Eq. 1 which passes inward through the inner surface is proportional to the angle which that surface subtends at r. This light is, for an infinitely long cylinder,

$$dL = 2\rho \sin^{-1}(a/r)\, r dr \qquad (2)$$

and from this the intensity of the light, Φ, at the inner surface can be obtained

$$\Phi = \frac{1}{2\pi a}\int_a^b dL = \frac{\rho}{\pi a}\int_a^b \sin^{-1}\left(\frac{a}{r}\right) r dr = \rho a f(a/b) \qquad (3)$$

where

$$f(a/b) = \frac{b}{2\pi a}\left[(b/a)\sin^{-1}(a/b) + \sqrt{1-\frac{a^2}{b^2}}\right] - \frac{1}{4} \qquad (4)$$

In Eq. 4, only the ratio of a and b appears, thus $f(a/b)$ can be plotted against the ratio a/b as shown in Fig. 3. To obtain a value of light intensity, Φ, for given values of a and b, one need only determine the value of $f(a/b)$ from Fig. 3 and multiply this by the inner radius, a, and by the amount of light generated per unit volume, ρ. For example, if $a = 3$ and $b = 4$, $\Phi = 0.39\rho$.

In an actual scintillator, the amount of light generated per unit volume will not be a constant throughout the light-generating sleeve because of the absorption of gammas within the sleeve. The actual output of the phototube will depend upon some average density of light generated within the sleeve.

Since the light passing through the inner surface of the luminescent sleeve comes from all directions, it is essentially diffuse. To simplify the problem of how much light reaches the photocathode, it is assumed that this light is perfectly diffuse. Then, for a very long cylindrical tube, the fraction, g, of the inward-bound light from any element of surface which strikes the photocathode is found to be (by reasoning similar to that which precedes Eq. 2)

$$g = \frac{2\sin^{-1}(d/2a)}{\pi} \qquad (5)$$

where d is the diameter of the photocathode.

If end effects are neglected, this same fraction holds for all the light going inward from the sleeve. The total light, L, striking the cathode is

$$L = (2\pi a z_c \Phi)\left[\frac{2\sin^{-1}(d/2a)}{\pi}\right] = 4az_c\Phi \sin^{-1}(d/2a) \qquad (6)$$

where z_c is the photocathode length.

Output Current and Voltage

If the sensitivity of the photosurface is γ amperes per watt, the current, I, produced by the light, L, is $I = \gamma L = 4\gamma a z_c \Phi \sin^{-1}(d/2a)$, and using Eq. 3 for Φ, and neglecting end effects

$$I = 4\gamma a^2 z_c \rho f(a/b) \sin^{-1}(d/2a) \qquad (7)$$

This is the expression for the total photocurrent in a long phototube in terms of the geometry of the phototube and sleeve, the photosensitivity, and the light output of the luminescent sleeve.

The characteristic impedance, Z, into which the current flows is given

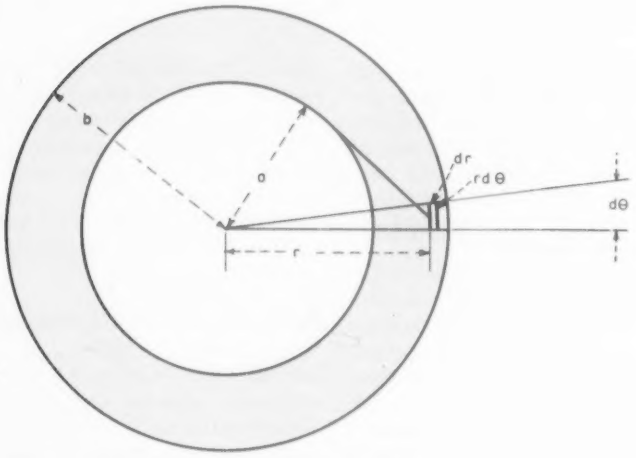

FIG. 2. Geometry of the light-generating sleeve

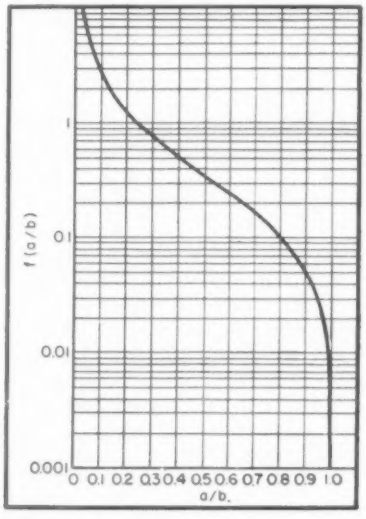

FIG. 3. Plot of $f(a/b)$ vs a/b

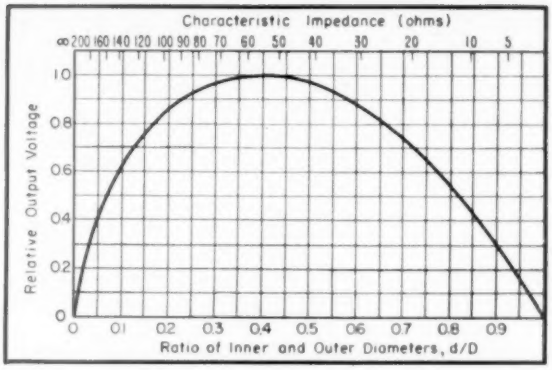
Characteristic Impedance (ohms)
∞ 200 160 140 120 100 90 80 70 60 50 40 30 20 10 5
Relative Output Voltage
0 0.2 0.4 0.6 0.8 1.0
0 0.1 0.2 0.3 0.4 0.5 0.6 0.7 0.8 0.9
Ratio of Inner and Outer Diameters, d/D

FIG. 4. Dependence of output voltage of coaxial phototube on ratio of inner and outer diameters for $D \approx 2a$

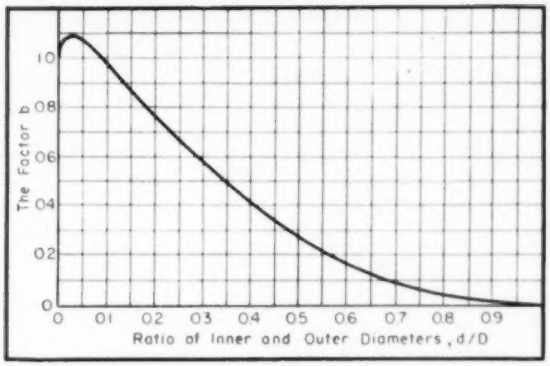

FIG. 5. The factor b in Eq. 11

by $Z = 60 \ln(D/d)$, where D is the diameter of the anode. Hence, the corresponding voltage is

$$E = IZ = 240\gamma a^2 z_c \rho f(a/b) \times \sin^{-1}(d/2a) \ln(D/d) \quad (8)$$

This is the voltage which would be developed by a coaxial phototube in which end effects are negligible. To take care of the case when end effects are important, the factor F is introduced. This factor is the ratio of the light flux emanating from a sleeve of finite length and striking a photocathode to the light flux which would strike a photocathode of the same length if it were an infinitely long sleeve. This factor depends upon the geometry of the phototube and surrounding sleeve, and can be anywhere in the range $0 \leq F \leq 1$. The final expression for output voltage, taking into account end effects, is

$$E = 240F\gamma a^2 z_c \rho f(a/b) \times \sin^{-1}(d/2a) \ln(D/d) \quad (9)$$

Optimum Impedance

The characteristic impedance which gives maximum output voltage can be found by maximizing Eq. 9 for variation in d, which gives

$$(d/2a) \ln(D/d) = \sqrt{1 - (d/2a)^2} \sin^{-1}(d/2a) \quad (10)$$

and by determining the relation between D and $2a$ for a given size scintillator sleeve. The values of D and $2a$ differ because of the thickness of the glass wall of the phototube and the spacing between the anode and the wall. For example, if $D = 8.5$ cm and $a = 5$ cm, $2a = D \cdot 0.85$. Substituting this value for $2a$ in Eq. 10 and solving graphically, it is found that, for $d = 4.5$ cm, the maximum output voltage occurs when $d/D = 0.38$, which corresponds to an impedance of 58 ohms. If the spacing between the inner surface of the sleeve and the anode is neglected, i.e., assume $D = 2a$, the optimum condition is $d/D = 0.39$, which corresponds to an impedance of 57 ohms.

The dependence of output voltage on the ratio of inner and outer diameters, assuming $D = 2a$, is shown in Fig. 4. It can be seen that an impedance in the range 40–80 ohms will give 95%, or more, of maximum output. A characteristic impedance of 50 ohms is a good choice because it gives 99% of maximum possible output voltage and is easily matched to many commercially available coaxial transmission lines to avoid multiple reflections.

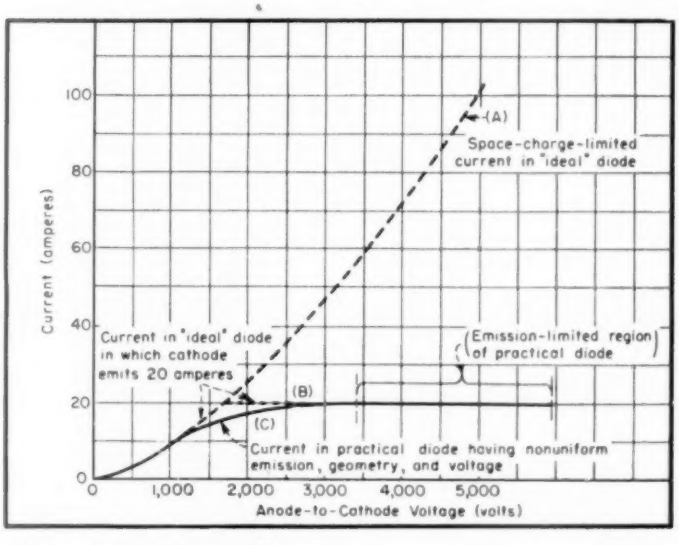
Current (amperes)
0 20 40 60 80 100
0 1,000 2,000 3,000 4,000 5,000
Anode-to-Cathode Voltage (volts)
(A)
Space-charge-limited current in "ideal" diode
Current in "ideal" diode in which cathode emits 20 amperes
(Emission-limited region of practical diode)
(B)
(C)
Current in practical diode having nonuniform emission, geometry, and voltage

FIG. 6. Current in coaxial diode

Space-Charge Effects

It has been tacitly assumed that the current which flows in a phototube depends only upon the light flux incident upon the photocathode. Under conditions of high current, however, the retarding effect of the space charge between cathode and anode places an upper limit upon this current. The maximum current, I_m, which can flow from any small section of the photocathode is proportional to the $\frac{3}{2}$-power of the voltage between that section and the anode, just as for any other similar diode. For long concentric cylindrical conductors, this maximum current is (2)

$$I_m = 1.468 \times 10^{-6} \frac{z_c E^{\frac{3}{2}}}{br} \quad (11)$$

where z_c = length of conductors, E =

anode-to-cathode voltage, r = anode radius, and b = a factor which depends on the ratio of inner and outer conductors as shown in Fig. 5.

In the derivation of Eq. 11, it was assumed that: (a) the surfaces of the anode and cathode are equipotential surfaces, (b) there are no gas molecules, (c) the emitted electrons have zero initial velocity, and (d) geometry is perfect.

For a coaxial phototube with z_c = 20 cm, r = 4.25 cm, and b = 0.24, Eq. 11 reduces to

$$I_m = 2.9 \times 10^{-4} E^{3/2} \qquad (12)$$

giving curve A in Fig. 6. In an "ideal" diode, having a uniform emission of 20 amperes, the current would follow curves A and B in Fig. 6 with a sharp "knee" at the intersection of the two curves. In a practical diode however, the nonuniform emission over the cathode surface and other irregularities will bring about localized space-charge limitations with the result that the current will follow a relationship somewhat like that of curve C. The exact relationship for a particular diode would have to be determined experimentally. Uniform illumination of the photocathode and uniform photosensitivity are important when the photocurrent is to be emission-limited, which is desired when the output current is to be proportional to the incident light flux.

The direct method of determining whether space-charge effects are important in a particular diode is to vary the anode-to-cathode voltage while holding the emission current constant at the desired value. If the output current is independent of anode voltage, emission limitation prevails.

Space charge also affects the transit time, T, of the photoelectrons in the phototube. Neglecting end effects and initial velocity of the electrons

$$T = \frac{K_1(D - d)/2}{5.95 \times 10^7 \sqrt{E}} \qquad (13)$$

where E is the anode-to-cathode voltage, D and d are as defined in Fig. 1, and K_1 is a factor which depends upon the space charge and the relative diameters of cathode and anode (*3*). For a typical tube, operated at 4,000 volts, transit time under space-charge-limited conditions is 1.4×10^{-9} sec; with no space charge it is 1.0×10^{-9} sec.

Scintillator Sleeve

For the design of the sleeve it was necessary to study the properties of scintillators which could be used, and the absorption of the generated light.

Characteristics of scintillators. A study was made (*4*) of the properties of crystal scintillators which could be of use for the luminescent sleeve. Before the discovery of liquid fluors, it had been decided on the basis of temperature, time, and efficiency measurements that stilbene was the optimum crystal, and varied very little in decay time or light output with temperature changes over the range of temperatures to be expected. Investigation of liquid fluors indicated even less variation of their properties. Accordingly, the scintillator used is a mixture of ½% terphenyl (p-diphenylbenzene) in toluene.

Measurements (*4*) based on gamma-ray, X-ray, and ultraviolet excitation yielded a value of 2.5×10^{-9} sec for the decay time of the scintillations, assuming a decay law of the form $e^{-t/\tau}$. The light output under constant gamma irradiation showed no change with temperature variation over the range of −90 to +80° C.

The saturation characteristics of a scintillating material are difficult to determine because of the high intensities at which saturation takes place. Exposure of a sample of liquid fluor to a strong radioactive source caused a noticeable change in the quantum

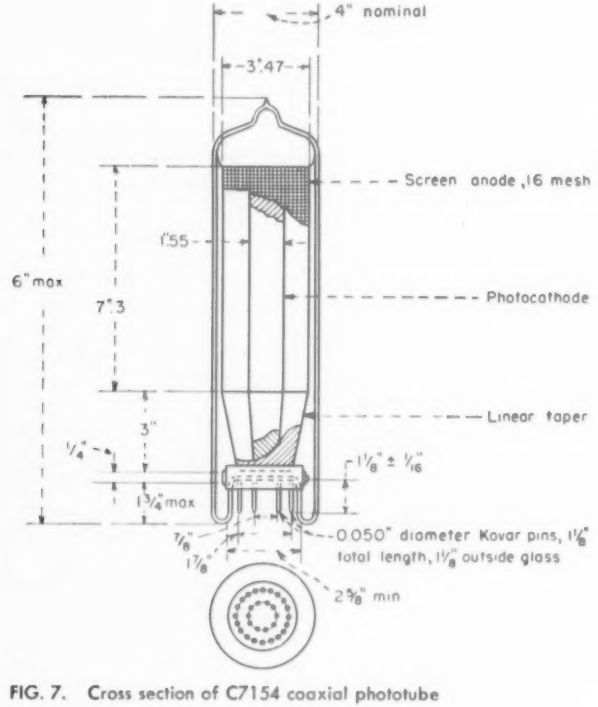

FIG. 7. Cross section of C7154 coaxial phototube

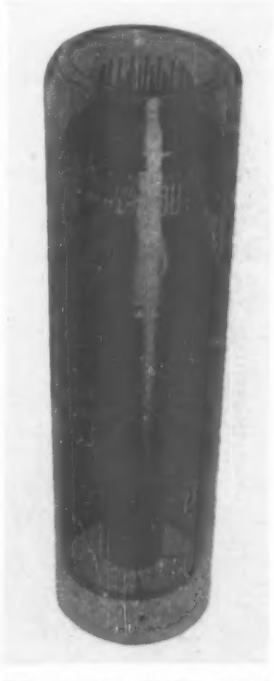

FIG. 8. The C7154 coaxial phototube

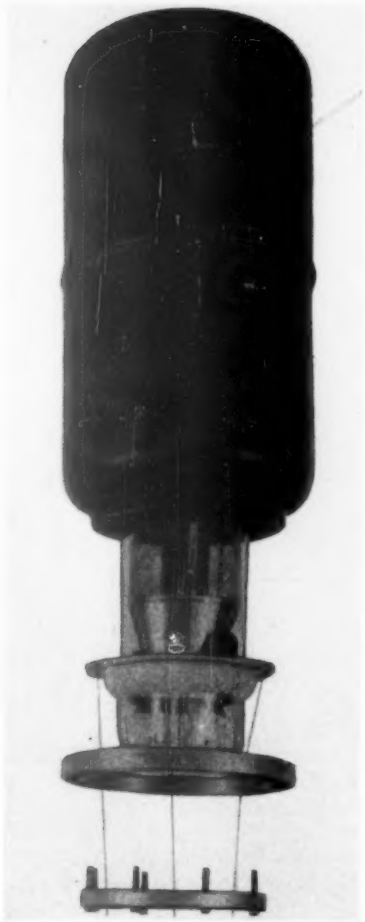

FIG. 9. Exploded view of liquid container and phototube

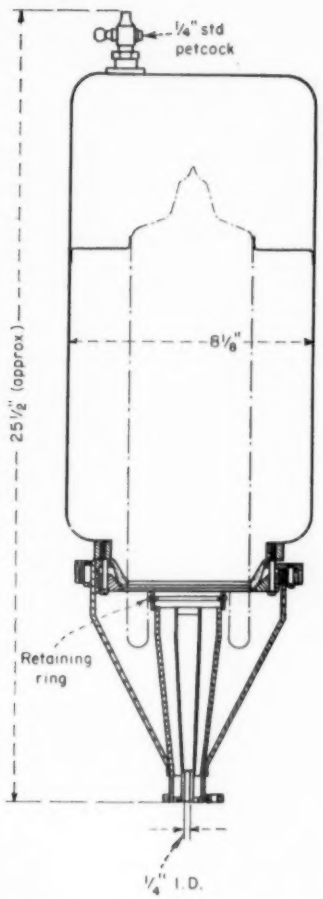

FIG. 10. Assembly of liquid container and socket

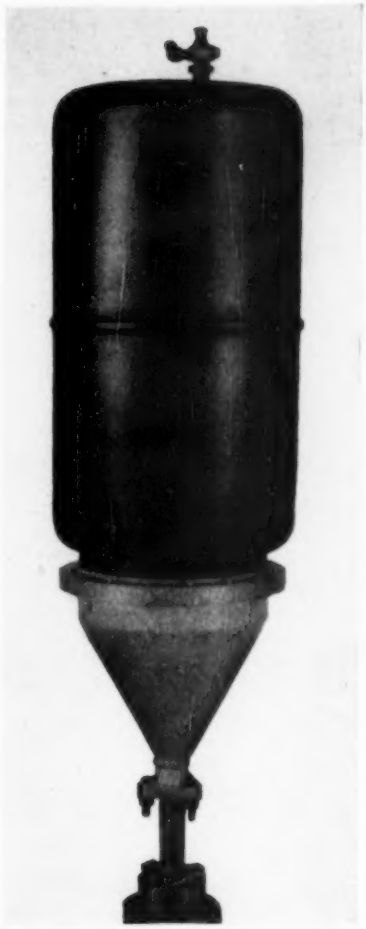

FIG. 11. Complete scintillation detector in upright position

efficiency. The total exposure was such that deterioration of fluorescent material should always be expected for high-intensity measurements.

Light absorption in scintillator sleeve. In addition to the geometrical factors which affect the sensitivity, the absorption of the generated light affects the detector sensitivity in accordance with sleeve thickness. The light absorptivity of the organic scintillators depends upon the wavelength of the light. Since these scintillators emit a fairly broad band of wavelengths, an effective absorptivity which is representative of the entire emission band is needed. For light of one wavelength attenuated by a thin slab of thickness x, it is possible to use the approximation

$$\Phi_\lambda = \Phi_{0\lambda} e^{-\alpha x} \approx \Phi_{0\lambda}(1 - \alpha x); \qquad \alpha x < 1 \quad (14)$$

for the transmitted intensity of light of wavelength λ, for which the absorptivity of the material is α. With a continuous distribution of wavelengths, the transmitted intensity becomes

$$\Phi_t = \int_\lambda I_{0\lambda} d\lambda (1 - \bar{\alpha} x) \qquad (15)$$

where $\bar{\alpha}$ is the weighted arithmetic mean of the α's and $I_{0\lambda}$ is the weighting factor. By analogy with Eq. 14, $\bar{\alpha}$ is the effective absorptivity of the material for the whole polychromatic beam.

The effective absorptivity, $\bar{\alpha}$, has been obtained for ½% terphenyl in toluene. The value found is 0.03 cm^{-1}.

Production Phototubes

A cross section of the production phototube is shown in Fig. 7, and a photograph in Fig. 8. The tube has a nickel-mesh anode and a photocathode with an S-4 surface on a nickel base. The envelope is Corning Glass 705-2.

A comparison of the transmission of the glass envelope with the emission of ½% terphenyl in toluene shows that the glass envelope attenuates the light output from this solution and therefore reduces the photocell sensitivity. The average photosensitivity of 22 of the first production phototubes, measured at the 3,650-Å line of mercury, was 1.9×10^{-2} amperes per watt.

A liquid container and tube socket have been designed as a single unit to hold a scintillating liquid around the coaxial phototube and to match it into a ⅞-in. copper coaxial cable. An exploded view is shown in Fig. 9, and an assembly drawing is given in Fig. 10. The complete detector with socket is shown in Fig. 11, and an exploded view of the socket is shown in Fig. 12. The tapered section of the socket has been designed to maintain a characteristic impedance of 50 ohms.

The phototubes have been desig-

nated RCA type C7154. Each phototube was given tests to determine: (a) its ability to deliver a peak current of 20 amperes into a 50-ohm line without space-charge limitation, (b) its ability to withstand, for long periods, an anode voltage sufficiently high to avoid space-charge effects, (c) its ability to furnish a high current pulse of several microseconds duration without initiating a voltage breakdown. In addition, several of the commercial phototubes have been tested more extensively to determine whether the photocurrent is linearly related to the intensity of the incident light.

Variation in Sensitivity

Since space-charge limitation in a diode depends upon variation in emission, an investigation of the variation of photosensitivity over the surface of the photocathode of the RCA C7154 phototubes was made by scanning the photocathode with a spot of light. A mercury vapor lamp with a combination of filters to isolate the 3,650-Å line was used as a light source. The 3,650-Å line was chosen because it is near the center of the emission spectrum of the scintillator material used. The results roughly confirmed the space-charge limitation effects.

Maximum Output Currents

The peak of the current pulse from a type 935 phototube was plotted against light intensity, and the linear relationship, with current densities as high as 0.1 ampere/cm², demonstrated the feasibility of drawing 20 amperes from a 200-cm² photosurface.

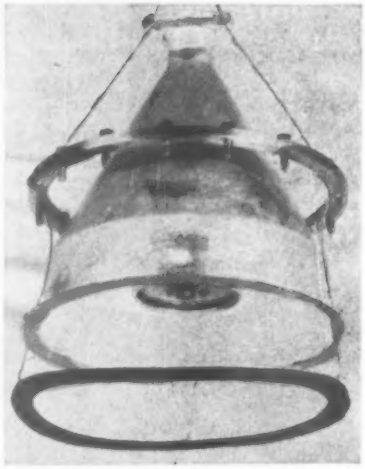

FIG. 12. Exploded view of tube socket

When the final C7154 model design was adopted, it was necessary to find the maximum current which could be delivered with 4,000 volts across the electrodes and a 50-ohm load in series with the tube. By plotting Child's law for the geometry of the C7154 and drawing a 50-ohm load line it can be seen that the maximum theoretical current that could be drawn would be about 27 amperes with 4,000 volts on the system. The end effects and nonuniformity of the production phototubes were such that at 4,000 volts and 18 amperes some space-charge effects were evident. Figure 13 gives a typical example of the space-charge limitation phenomena.

The tests performed on the C7154 phototubes to determine space-charge effects were of two kinds. One method was to illuminate the tube with uniform light pulses of a given amplitude and vary the voltage on the phototube while readings were made of the pulse of photocurrent. The other method was to apply a given voltage while the light intensity was varied. Figure 14, a typical plot of photo-current versus light intensity, shows the phototube to be linear in the region in which space-charge effects are unimportant.

For every C7154 phototube received, space-charge data similar to that of Fig. 13 were obtained. If a tube is operated at a voltage that establishes the operating point just over the knee of the space-charge curve, the transit time of some of the electrons can be appreciable; therefore, to insure good response time, the voltage should be 1,000 volts above the voltage corresponding to the upper end of the knee.

Voltage Breakdown

Two types of voltage breakdown have been observed. One was a discharge across an insulator in the cell;

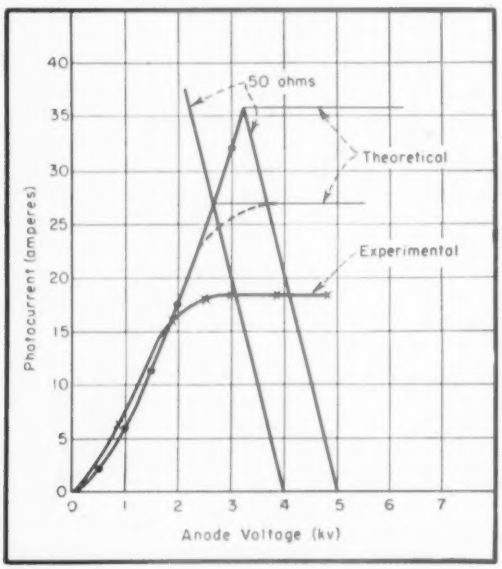

FIG. 13. Typical space-charge characteristics of production coaxial phototubes

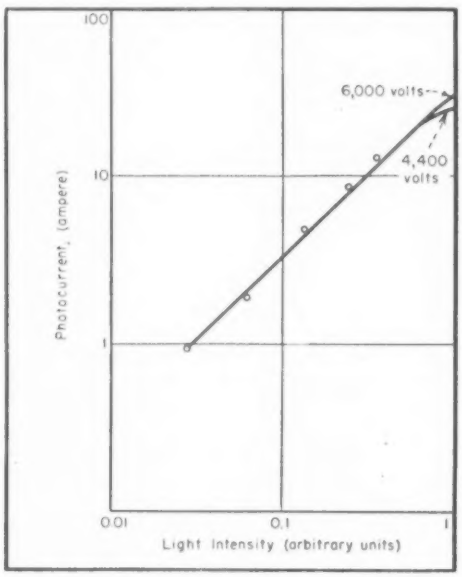

FIG. 14. Linearity of phototube exposed to high-intensity light flashes

the other, found in nearly all the developmental models, was a gas breakdown. This latter type of breakdown could be slightly improved by repeated breakdowns, but over a period of weeks the breakdown voltage was found to decrease.

The first type of breakdown occurred mainly in the final production tubes. This type of breakdown consisted of a breakdown along a glass path, probably due to a deposit of cesium. A typical tube would break down in this fashion first at 5,000 volts, then at about 5,500, 6,500, 7,000, 8,000, and 9,000 volts. It seemed that after each breakdown there would be some sort of clearing of the path so that the breakdown potential was increased. None of the production models of the C7154 experienced gas breakdowns; all breakdowns were the localized type. There were only a few phototubes which did not clear themselves with repeated breakdowns and most of them became usable at 6,500 volts.

Neither of the two types of breakdown seemed to have damaged the photosensitivity of the photosurface. Because of the heat generated, it is probable, however, that many rapidly recurring breakdowns would damage the photosurface.

It was found that both types of breakdown could be initiated by light flashes. A tube in the dark might withstand 7,000 volts without a breakdown, but when a bright source of light was flashed upon the cell, a breakdown would take place from 10 to 100 μsec following the photocurrent associated with the light flash. This type of breakdown was also reduced by repeated breakdown.

The tubes selected for use were chosen from those which did not break down at 6,500 volts with intense light pulses, even though a delayed breakdown (10–100 μsec after the initiating light flash) would make it possible to use such a tube and still accurately record the initial part of a light flash.

Decreasing the impedance of the phototube by increasing the diameter of the photocathode has the effect that less anode voltage is required for a fixed output voltage. This lowering of anode voltage lessens the problem of high-voltage breakdown. However, a 50-ohm characteristic impedance was chosen for the phototube to reduce reflections at the connection to the 50-ohm transmission line.

Fatigue of Photosurface

It was necessary to determine whether any fatigue of the surface of the photocathode could be expected, since it is known that photosurfaces do suffer fatigue effects when operated at high levels for long periods. This was investigated by flashing a large pulse of light of about 20 μsec duration, and then, during this pulse, to flash still another pulse of 1 μsec duration. Under these conditions no fatigue was observed.

The temperature dependence of the response of the C7154 was another factor that had to be investigated to determine the photosensitivity under different temperature conditions. To do this, the tube was set up in a cool room with light flashes being produced every 10 sec. The amplitude of these flashes was recorded about every 10 min as the temperature was raised from 30 to 51° C. A 250-watt infrared heat lamp, with the visible part of the spectrum cut out by a thin sheet of gelatin filter paper, was used for heating the photosurface. The temperature was read with a thermocouple soldered directly to the pins of the photocathode. No significant change in pulse heights of the phototube output occurred during the entire experiment.

Effect of Shield

Since the phototubes are used surrounded by a metal can at ground potential, it was necessary to determine what effect a metal shield has upon the electrodynamic characteristics of the phototube. A copper screen was wrapped around a phototube and potentials ranging from +1,000 volts to −1,000 volts were applied to it while various levels of light pulses were observed. The shield potential between these levels had no apparent effect upon the pulse shape or pulse size.

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Which Counter for Carbon-14?

THE COUNTING of carbon-14 may be done by any of the standard methods—electroscope, G-M counter, or proportional counter, either with the thin-window or the windowless type . . . The low energy of the C^{14} beta rays heavily weight the choice for the windowless counters if low specific activities are to be measured. Unfortunately, nearly all C^{14} work seems to lead to analysis of low-activity products.

Many people prefer the more expensive and complicated windowless counters, although there is much to be said for the less costly end-window counting set-up . . . For the price of one complex array, two end-window set-ups may be obtained—one on a standby basis for use during the periods of repair of the regularly-used instrument. But, again, both types of counters have their loyal supporters.

There is one difficulty that has been experienced in using windowless counters for barium carbonate or any other sample material that is a poor electrical conductor. The samples tend to pick up a variable amount of charge both in storage and in the counting chamber. This produces a variation in the count measured at different times that exceeds the normal statistics of counting. In addition, the charge build-up on the sample in the counting chamber leads to counting rates that diminish with the length of time the sample is in the chamber. This can be combatted by adding graphite to the sample, spattering with metal, placing a coarse screen over the sample, and at least partially by the use of metallic sample mounts.

—D. R. SMITH, *Oak Ridge Institute of Nuclear Studies, August 25, 1952*

Studies on

EDTA Treatment of Internal Radioactive Contamination

Tests with rats injected with yttrium-91 show that fission products can be removed from the body either before or after fixation by the bone. The work also suggests the feasibility of a prophylactic approach to preventing skeletal deposition of fission products

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THE RADIOLOGICAL SAFETY of a population exposed to nuclear fission, whether in industry or warfare, involves prevention of injuries resulting from the internal deposition of fission products and fissile material.

Of the radioisotopes produced in fission, a considerable portion have a long biological half-life and present a chronic toxicity hazard once they enter an organism. Strontium-89,90, yttrium-91, cerium-144 and zirconium-95 are examples of these nuclides. The fissile material itself may also be present and, in the case of plutonium-239, has a much longer biological half-life than the nuclides mentioned.

On entering the body, the isotopes named are rapidly removed from the blood stream, and those fractions not excreted are fixed almost entirely by the skeleton. They remain in the skeleton for a long period of time, since they are eliminated at a very slow rate and have a long half-life (*1*).

The deposition of minute amounts of these radioactive materials in bone leads to serious pathological changes in the irradiated tissue and indirect changes elsewhere because of the long period of irradiation to a localized area of bone and particularly to the radiosensitive bone marrow.

The problem, then, is essentially that of (1) removing these internally deposited nuclides from the body as rapidly as possible, inasmuch as there is, at present, no proven way of counteracting their radiation effects, or (2) if possible, preventing their deposition. Studies on both parts of this problem are reported here.

Y^{91}, in itself one of the important fission products (*2*), was selected for

TABLE 1—Time Interval Between Na-EDTA and Ca-EDTA Administration After a Single Injection of Yttrium-91

Days after injection of Y^{91}	*Time interval between injections*
1–14	no treatment
15	3 hr
16	2 hr
17	1 hr
18	0 (given simultaneously)
19–21	no treatment
22	3 hr
23	2 hr
24	1 hr
25	0 (given simultaneously)
26–28	no treatment
29	sacrifice

both studies because it is a prototype of the lanthanide rare earths which constitute a considerable portion of the fission yield. Others have shown (*3*, *4*, *5*) that Y^{91} is removed rapidly from the blood stream on absorption from the lungs, gastro-intestinal tract, etc., and is then deposited chiefly in bone, where it has a biological half-life longer than 500 days (*3*).

1. REMOVAL OF BONE-DEPOSITED FISSION PRODUCT

The classical procedures for removing bone-deposited radium, such as the use of a low calcium diet, and the administration of agents designed to cause bone resorption (parathormone, ammonium chloride and citrate), have been shown to have little effect in removing the lanthanide rare earths (*1*, *6*).

Schubert (*7*) has reported an increased excretion and a consequent lowered deposition of yttrium in bones of rats following administration of multiple massive dose injections of zirconium citrate, when the latter treatment was administered starting one hour after intraperitoneal injection of Y^{91}.

Foreman, in a recent study (*8*), employed a chelating agent to accelerate the excretion of radioelements from rats. He reported a marked enhancement of urinary excretion of Y^{91} and P^{239} following administration of calcium EDTA (ethylene diamine tetra acetic acid) and Fe-3 (a Versene complex). The calcium EDTA (Ca-EDTA) has been shown to exchange calcium for other metals *in vivo*, forming a soluble complex which is excreted chiefly in the urine (*8*, *9*). Rubin (*10*) reports that following the administration of Ca-EDTA to a patient having acute lead poisoning, clinical improvement was observed along with an increase in the urinary excretion of lead. This is probably the first clinical use of EDTA in a case of metallic poisoning, and emphasizes the fact that Ca-EDTA is apparently nontoxic to humans at a therapeutic dose level.

While the use of Ca-EDTA has been shown to have considerable effect in increasing the excretion of yttrium, plutonium, and lead, the source of these metals has never been shown clearly to be bone and in the cases mentioned probably comes in large part from the soft tissues and blood. This is certainly true in the instances when EDTA (or other) treatment was started within hours after internal contamination. The effect of the Ca-EDTA appears to be limited when these elements are deposited in bone.

For the removal of bone-fixed nuclides, more rigorous measures are apparently required. Rubin has therefore suggested the use of Ca-EDTA and sodium EDTA (Na-EDTA) to effect the removal of bone-deposited lead and other metals (*10*). In the work reported here, the technique of giving multiple injections of both Na-EDTA and, followed at short intervals, Ca-EDTA, was employed to bring about the removal of bone-deposited yttrium.

Experiments

Two experiments were performed, differing only in the frequency with which EDTA injections were given. In each of these experiments 20 Sprague-Dawley male rats (200 to 277 gm in weight) were injected intraperitoneally with 25 μc of carrier-free $Y^{91}Cl_3$. The animals in each experiment were then separated into two equal groups, one of which underwent EDTA treatment for the removal of Y^{91}, and the other of which served as a control.

All the animals were placed in individual metabolism cages and fed *ad lib.* throughout the experimental period with Purina Laboratory Chow. The metabolism cages were suspended in large glass jars which were lined with plastic bags. The bottoms of the cages were coated with paraffin to facilitate recovery of the excreta. All the excreta were collected in the bags, which were then ashed directly, to insure against the loss of any radioactivity, and to minimize the hazard to personnel in the preparation of excreta for analysis.

After a period of 14 days following intraperitoneal injection of rats with Y^{91}, the excretion of yttrium dropped to a minimum value, and the Y^{91} remaining in the animals can be assumed to have been fixed, chiefly in the skeleton (*3*, *4*, *5*, *6*).

Experiment A. This test consisted of a daily intraperitoneal injection of 10 mg of tetrasodium EDTA (pH = 8.4), followed at various time intervals by the injection of 50 mg of calcium disodium EDTA (pH = 7.4). Table 1 indicates the intervals used between the EDTA injections.

Experiment B. The pairs of injections were administered twice daily, with all other conditions the same.

In both experiments, the injections were given on eight days during a two-week period. During this two-week experimental period, the excreta were collected from the animals in both the treated and control groups. Both groups of animals were sacrificed on the 29th day post-Y^{91}-injection.

Both tibiae were removed from each animal after sacrifice and were analyzed independently in the following manner. The bones were dissected free of muscle, dried at 100° C, and weighed. The tibiae were then dry ashed at 600° C overnight, weighed, and dissolved in 2*N* HCl. The solution was made up to 25 cm^3 and aliquots were taken for radioactive assay. Samples were then dried in metal cups, and the radioactivity determined by means of an end-window Geiger counter and standard scaling circuit. Radioactivity is expressed in per cent of a standard made up from the injection solution. To compensate for mass absorption effects, an amount of ash equivalent to that found in the experimental samples was added to the injection standards.

In both experiments, the excreta for each animal were collected and pooled at the end of the first 14 days (pre-treatment period), and again at the end of the experiment (post-treatment period). The cages were then washed with dilute HCl for more complete recovery of the activity.

The excreta and washings were ashed directly in the plastic bags, and the ash dissolved in 1*N* HCl, made up to a suitable volume, and an aliquot analyzed for radioactive content.

Results and Discussion

The administration of Na-EDTA followed by Ca-EDTA to rats which had been injected two weeks previously with tracer amounts of Y^{91} caused a marked increase in the total excretion of the isotope (see Tables 2 and 3) compared with the corresponding excretion of the controls. The tables show that the mean total excretion of Y^{91} for the 2-week pre-EDTA-treatment period in both the control and the experimental animals in both experi-

TABLE 2—Distribution and Excretion of Yttrium-91 for Rats Treated Daily With Na-EDTA Followed by Ca-EDTA (Expt. A)

Rat No.	Final body weight (gm)		Tibia ash weight (mg)		Tibia length (mm)		Units† tibia P'tase per gram ash		Average % Y^{91} per tibia‡		% Y^{91} in skeleton‡,§		% Y^{91} excreted before EDTA treatment‡		% Y^{91} excreted after EDTA treatment‡	
	C*	X*	C	X	C	X	C	X	C	X	C	X	C	X	C	X
1	247	243	248	225	38.5	37.0	2.37	2.73	2.22	1.77	58.7	48.1	36.7	41.5	4.8	14.4
2	213	265	213	238	39.0	37.5	2.27	2.36	2.47	1.88	66.7	50.7	30.8	36.8	4.0	14.2
3	266	237	262	204	37.0	39.0	2.37	3.29	2.26	1.72	61.4	45.4	29.1	38.2	6.1	16.6
4	242	207	272	206	36.5	38.0	2.76	3.01	2.13	1.71	56.1	45.4	30.6	27.6	4.5	16.3
5	253	251	256	249	38.5	39.0	2.92	2.60	2.00	1.97	53.4	53.4	27.1	30.9	6.2	15.5
6	230	243	211	250	38.0	38.0	2.57	3.13	2.28	1.89	61.4	50.7	29.5	34.4	4.6	17.3
7	242	249	248	234	39.0	38.5	3.15	2.47	2.60	1.86	69.4	50.7	51.0	38.8	3.7	18.0
8	230	241	249	227	37.0	38.5	2.62	2.81	2.55	1.69	69.4	45.4	37.7	38.0	3.7	19.3
9	260	243	264	225	37.0	39.0	3.86	4.17	2.41	1.49	64.1	40.0	31.2	28.8	3.7	24.3
10	247	229	239	232	37.0	38.0	3.94	3.53	2.26	1.76	61.4	48.1	33.5	39.6	3.2	16.2
$\bar{X}$*	243	241	246	229	37.8	38.3	2.88	3.01	2.32	1.77	62.2	47.8	33.7	35.5	4.5	17.2
σ*							0.62	0.56	0.20	0.13			6.4	4.9	1.0	3.0

TABLE 3—Distribution and Excretion of Yttrium-91 for Rats Treated Twice Daily with Na-EDTA Followed by Ca-EDTA (Expt. B)

Rat No.	Initial body weight (gm)		Final body weight (gm)		Tibia ash weight (mg)		Average % Y^{91} per tibia		% Y^{91} in skeleton§		% Y^{91} excreted before EDTA treatment		% Y^{91} excreted after EDTA treatment		% Y^{91} in liver	
	C*	X*	C	X	C	X	C	X	C	X	C	X	C	X	C	X
1	230	223	239	208	238	220	2.12	1.22	56.6	32.6	33.6	55.2	3.96	10.6	1.49	0.14
2	227	249	250	246	239	270	1.75	1.24	46.7	33.1	40.6	57.0	3.15	10.8	1.66	0.65
3	230	258	246	262	251	273	2.01	1.12	53.7	29.9	44.0	60.8	2.56	9.8	2.07	0.31
4	237	232	235	244	246	252	1.94	1.51	51.8	40.3	51.3	44.6	3.60	14.7	0.84	0.75
5	249	200	270	178	274	232	2.18	1.77	58.2	47.3	40.5	32.0	2.56	22.7	1.57	0.44
6	246	277	230	272	274	288	2.36	1.53	63.0	40.9	38.9	39.5	3.81	17.8	1.13	0.53
7	229	249	248	261	268	277	2.35	1.51	62.4	40.3	40.0	39.0	3.73	18.2	1.52	0.40
8	234	226	267	245	276	245	2.28	1.48	60.9	39.5	26.8	38.7	4.60	21.5	1.97	0.26
9	231	234	244	237	263	245	1.86	1.69	49.7	45.1	33.9	37.7	5.50	17.7	....	0.47
10	231	243	248	238	250	262	2.04	1.67	54.5	44.6	38.2	35.7	3.32	18.4	1.07	0.50
$\bar{X}$*	234	239	248	239	258	256	2.09	1.47	55.8	39.4	38.8	44.0	3.68	16.2	1.48	0.45
σ*							0.17	0.19			6.7	9.8	0.98	4.7	0.44	0.20

* C = control; X = EDTA-treated; $\bar{X}$ = mean; σ = standard deviation.
† King and Armstrong unit (mg phenol liberated from disodium phenyl phosphate in 30 min at 37° C and pH = 10.1).
‡ Represents per cent of injected dose, adjusted to 100% recovery. Actual recovered dose was 97.2% of injected dose.
§ Calculation based on ratio of total skeletal Y^{91} content/tibia Y^{91} content = 26.7.

ments was about 38% of the injected dose.

After treatment with the EDTA, the experimental animals in experiment A excreted a mean of 17% of the injected Y^{91}, the controls only 4.5%. In experiment B, following twice-daily treatments, the corresponding values averaged 16% for the treated animals, and 3.7% for the controls.

Thus, analysis of the excreta for the 2-week period of EDTA treatment in both experiments indicates an excretion of Y^{91} approximately four times greater for the treated animals than for the untreated controls.

Examination of the tibiae from the control and treated groups of both experiments reveals a marked decrease in the Y^{91} content of the tibiae of the EDTA-treated animals. In experiment A, for example, the tibiae of the treated animals contained a mean of 1.8% of the injected dose, while the control animals averaged 2.3%. Thus, in relative amounts, the tibiae of the treated animals had only about 76% as much Y^{91} as the control tibiae.

After twice-daily treatment (experiment B), the tibia of treated animals contained about 70% of the Y^{91} content of the controls. The mean tibia Y^{91} content was not significantly different following twice-daily administration of the EDTA from that resulting from once-daily treatment.

The mean ratio of total skeletal Y^{91} content to the Y^{91} content of one tibia after intraperitoneal injection of Y^{91} was found to be 26.7. This was determined by experimental means, i.e., both tibiae and the remainder of the skeleton were analyzed separately. An estimate of the amount of the Y^{91} in the whole skeleton can be obtained from the amount found in the tibia by the use of this factor. The results of this calculation are also shown in Tables 2 and 3. In the control groups, a mean value of 59% of the injected dose of Y^{91} was calculated to be present in the whole skeleton 2 weeks after Y^{91} injection. This value is in good agreement with the results of Hamilton

(*3*), who found a retention of 63% under similar conditions.

The only other major organ in which the yttrium is stored is the liver, which contains about 1.5% of the injected dose at 29 days following intraperitoneal injection of Y^{91}. Muscle, which constitutes 45% of the body weight of rats, contains about 2% of the injected dose of yttrium 3 days after intraperitoneal injection, and only a fraction of this amount is found at 29 days after injection (*11*).

The increased amount of Y^{91} found in the excreta of the EDTA-treated animals correlates quite closely with the decreased skeletal Y^{91} content. For example, in experiment A, the decrease in Y^{91} content of the skeleton of the EDTA-treated animals, in terms of the actual amount of Y^{91} in the skeleton, as compared with their controls, was 15% of the injected dose. The absolute increased Y^{91} content of the excreta of the treated rats over that of the controls was 13% of the injected dose. In experiment B, the enhanced Y^{91} excretion in absolute amounts was 13% of the injected dose; 16% of the injected dose of Y^{91} was apparently removed from the skeleton of the treated animals. Thus, in absolute quantities, the enhanced excretion of Y^{91} from the body appears to be accounted for by the removal of the Y^{91} fixed by the skeletal tissue.

Tables 2 and 3 list the weight and length of the tibiae of the control and the treated animals. From the results of the two experiments, the treated animals do not differ significantly from the controls in body weight, tibia weight or tibia length. Thus, there appears to be no significant decalcification of the skeleton resulting from administration of Na-EDTA. It is interesting to note that the level in the tibia of the enzyme alkaline phosphatase, which is intimately related to bone metabolism, was not affected by EDTA treatment, as carried out in this study (see Table 2).

The action of the EDTA may be explained in the following manner. Administration of the tetrasodium salt results in a lowering of the blood calcium ion level through the formation of the Ca-EDTA complex. This, in turn, leads to a shift in the bone-blood calcium and nuclide equilibrium, resulting in the dissolution of bone. With the release of calcium from the bone, there is a simultaneous release into the blood of the bone-fixed yttrium. It is likely that both EDTA salts then combine with the bone-liberated yttrium in the blood and soft tissue. The injection of the Ca-EDTA probably acts as an adjuvant to the Na-EDTA by exchanging its calcium for the bone-liberated yttrium. An yttrium EDTA complex is formed since the EDTA salts have a stronger affinity for yttrium than for calcium ions. This soluble complex is presumably not metabolized by the body and is rapidly excreted in the urine.

Thus, the effectiveness of the removal of bone-deposited Y^{91} by the Na-EDTA plus Ca-EDTA treatment probably lies in the ability of the Na-EDTA to cause the dissolution of bone *in vivo*, and thus to make the bone-fixed Y^{91} available to EDTA in the blood stream, where it is then complexed. The amount of Na-EDTA that can be used is limited, however, as large amounts have proved toxic to rats (*9*). Ca-EDTA, on the other hand, is practically nontoxic to rats or human beings so far as is known (*8, 9, 10*). The combined use of both forms of EDTA, however, produces effective results without apparent harm.

2. PREVENTION OF FISSION-PRODUCT DEPOSITION

This investigation endeavored to determine the possibility of preventing the deposition of nuclides in the skeleton of rats by treating them with Na-EDTA followed by Ca-EDTA, either before or soon after a Y^{91} injection.

Experiments

In both experiments, 20 adult Sprague-Dawley male rats were used; 10 of these served as controls. The conditions for animal care, tissue removal, and excreta analysis were as described in the experiments in Part 1.

Experiment C. Treatment was initiated one hour after administration of Y^{91}; 25 μc of Y^{91} (as carrier-free $Y^{91}Cl_3$) were injected intraperitoneally in 20 rats. After one hour, 10 rats were injected intraperitoneally with a solution of 10 mg Na-EDTA (pH = 8.4). This treatment was followed, at an interval of 15 minutes, by the injection of 50 mg of Ca-EDTA (pH = 7.4). The two injections were administered twice daily for three days following injection of the Y^{91}.

A further part of this experiment was carried out, differing from that just mentioned only in the time at which the administration of EDTA was started. Intervals of ½ and 3 hours after Y^{91} injection were employed. Four rats were used in each experimental and control group.

After 5 days, all the animals were sacrificed. Both tibiae were removed and analyzed separately for their Y^{91} content. The liver, muscle, remains of the carcass, and the skeleton were analyzed for their radioactivity in half of the experimental rats and half of the control rats. The skeleton was separated from the carcass by boiling the animal in a strong soap solution for 90 minutes. All the tissues were dry ashed and made up to volume with 2*N* HCl solution. An aliquot was dried in a metal cup, and the radioactivity determined as before.

Experiment D. The salts of EDTA were administered before the injection of Y^{91} to prevent the deposition of the injected yttrium in the skeleton. An intraperitoneal injection of 10 mg of Na-EDTA was given to each of 10 of a group of 20 rats; 15 minutes later, 50 mg of Ca-EDTA were injected. One hour after the injection of the EDTA solutions, 25 μc of Y^{91} were administered intraperitoneally to all 20 rats. The animals were then placed in individual metabolism cages and their excreta collected.

In further tests, rats were pretreated with the salts of EDTA at time intervals of ½ and 3 hours before Y^{91} injection. Four rats were used in each experimental and control group.

All the animals were sacrificed three days after injection of the Y^{91}, and various tissues were removed for analysis. At the same time, the accumulated excreta of both experimental and control animals were analyzed for their radioactive content.

Results and Discussion

Table 4 presents those results of experiment C. The mean total Y^{91} excretion of the control group at the end of the 5-day experimental period was 29% of the injected dose; the mean total excretion of the EDTA-treated animals was 80% of the in-

jected dose. Thus the EDTA-treated animals excreted more than two and a half times the yttrium excreted by the untreated controls during the experimental period.

Table 4 shows also that at 5 days after administration of Y^{91} the tibiae of the EDTA-treated animals had a mean Y^{91} content of 0.67% of the injected dose, as compared with 2% for the controls. From these data, an estimate of the total skeletal Y^{91} content of the animals was derived, using the value of 26.7 as the ratio of Y^{91} contained in the whole skeleton to that contained in one tibia. Thus, at 5 days post-Y^{91}-injection, the mean total Y^{91} content of the skeletons of the control animals was 53% of the injected dose. Therefore, the skeletal tissue of the EDTA-treated animals contained only one third of the Y^{91} fixed by the controls.

A preliminary experiment on the uptake of radioyttrium by the skeleton showed that, following intraperitoneal injection, the Y^{91} concentration did not reach its approximate maximum value until six hours after the injection (Fig. 1). One hour after injection the Y^{91} content of the skeleton was approximately 25% of the injected dose.

TABLE 4—Distribution and Excretion of Injected Yttrium-91 for Rats Treated with Both Na-EDTA and Ca-EDTA One Hour After Injection (Expt. C)

Rat No.*	% Y^{91} in tibia		% Y^{91} in skeleton§		% Y^{91} in liver		% Y^{91} in skin and muscle		% Y^{91} in excreta¶	
	C‡	X‡	C	X	C	X	C	X	C	X
1	1.98	0.65	52.9	17.4	10.9	1.6	5.5	2.3	30.6	83.2
2	2.10	0.60	56.1	16.0	10.5	2.1	...	...	27.1	74.0
3	2.06	0.70	55.0	18.7	14.2	2.0	...	...	26.4	79.5
4	1.99	0.59	53.1	15.8	13.1	1.9	...	...	26.6	....
5	1.93	0.59	51.5	15.8	11.5	1.7	...	...	28.9	77.0
6	1.97	0.63	52.6	16.8	....	....	...	...	29.9	82.8
7	2.09	0.79	55.8	21.1	....	....	...	...	29.0	....
8	1.91	0.73	51.0	19.5	....	....	...	...	30.8	75.6
9	1.87	0.65	49.9	17.4	....	....	...	...	32.9	83.8
10	1.86	0.77	49.7	20.6	....	....	...	...	30.5	81.8
$\bar{X}$‡	1.89	0.67	52.9	17.9	12.0	1.9	5.5	2.3	29.3	79.7
σ‡	0.09	0.06	2.4	1.8	1.8	0.2	...	...	2.1	3.6

TABLE 5—Distribution and Excretion of Injected Yttrium-91 for Rats Pretreated with Both Na-EDTA and Ca-EDTA One Hour Prior to Injection (Expt. D)

1†	2.23	0.14	59.5	3.7	9.7	....	6.8	2.2	23.6	93.4
2	2.29	0.17	61.1	4.5	10.5	....	...	...	22.3	93.4
3	2.34	0.15	62.5	4.0	10.2	....	...	...	24.4	93.6
4	2.32	0.15	61.9	4.0	9.2	0.63	...	...	22.8	95.4
5	1.78	0.14	47.5	3.7	9.1	0.45	...	...	27.5	92.8
6	1.69	0.13	45.1	3.5	9.5	0.39	...	...	26.1	93.1
7	2.36	0.15	63.0	4.0	11.0	0.42	...	...	25.9	94.6
8	2.05	0.15	54.7	4.0	8.7	0.59	...	...	26.0	89.0
9	2.34	0.14	62.5	3.7	9.6	0.52	...	...	28.0	91.8
10	2.33	0.15	62.2	4.0	9.2	0.53	...	...	26.8	92.8
$\bar{X}$‡	2.17	0.15	58.0	3.9	9.7	0.50	6.8	2.2	25.3	93.0
σ‡	0.24	0.01	6.4	0.3	0.7	0.1	...	...	2.0	2.1

* All rats (Table 4 only) sacrificed 5 days post-Y^{91}-injection.
† All rats (Table 5 only) sacrificed 3 days post-Y^{91}-injection.
‡ C = control; X = EDTA-treated; $\bar{X}$ = mean; σ = standard deviation.
§ Calculation based on ratio of total skeletal Y^{91} content/tibia Y^{91} content = 26.7.
¶ Represents per cent of injected dose adjusted to 100% recovery. Actual recovered dose was 95.9% of injected dose.

In the first part of experiment C, EDTA treatment was begun one hour after the Y^{91} injection, by which time 25% of the injected dose must have been deposited in the skeletal tissue. Thus, the 50% increase in the yttrium excretion of the EDTA-treated rats could be derived only in part from the Y^{91} fixed by the skeleton. The remainder of the increased Y^{91} excretion was derived from the Y^{91} which had been rapidly complexed by EDTA in the blood and soft tissues. The yttrium-EDTA so formed was rapidly excreted from the body and consequently was prevented from being deposited in the skeleton.

A similar interpretation may be applied to Schubert's work (7), in which zirconium citrate treatment for the removal of injected Y^{91} was started one hour after the injection. Here also, a large portion of the increased Y^{91} excretion after zirconium treatment was probably derived from sources other than the skeleton.

Results of the experiments in which EDTA treatment was initiated ½, 1, and 3 hours after Y^{91} injection are presented in Fig. 2.** It can be seen that treatment started one-half hour post-Y^{91}-injection has apparently a greater effect in lowering the skeletal Y^{91} content than treatment started at 1 hour post-Y^{91}-injection. While EDTA treatment started 3 hours after injection of Y^{91} is not as effective as treatment started earlier, the skeletal Y^{91} is significantly lower in the EDTA-treated animals than in the corresponding controls. Hence, the earlier the EDTA treatment is started after the Y^{91} injection, the lower will be the final skeletal Y^{91} content.

Table 5 presents the results of experiment D, in which the EDTA was administered one hour prior to Y^{91} injection. The mean excretion of Y^{91} in the EDTA pretreated groups amounted to 93% of the injected dose compared with a mean of 25% in the control group. Of the 7% of the injected dose of Y^{91} which remained in the body of the EDTA-treated animals, approximately 4% was found in the skeleton, and the remaining 3% in the blood and

** Data for Fig. 2 were obtained from additional experiments which were of a preliminary nature and used smaller groups of rats. The experiments were carried out under the same conditions as the others reported here. It includes the 1-hr data from Tables 4 and 5. Control values represent the combined average for all the control animals used.

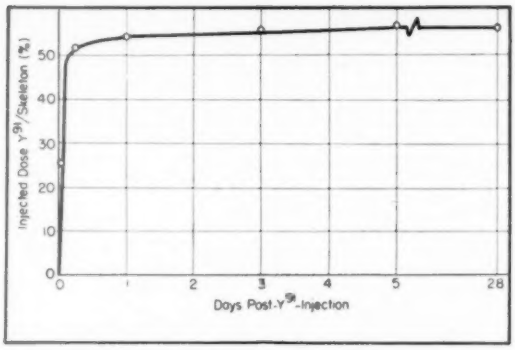

FIG. 1. Uptake of injected yttrium-91 by the skeleton of rats

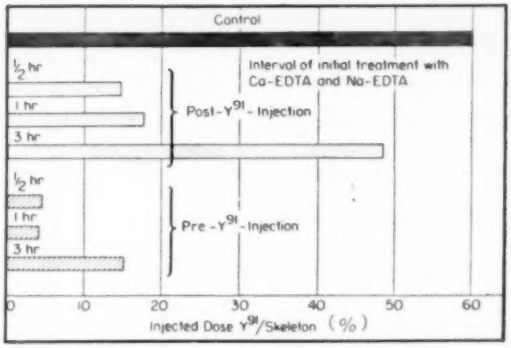

FIG. 2. Skeletal concentration of injected yttrium-91 in rats posttreated and pretreated with Na-EDTA and Ca-EDTA

soft tissues. The control group, on the other hand, retained 75% of the injected dose, two-thirds of which was found in the skeletal tissue. Thus, the treatment of rats with Na-EDTA and Ca-EDTA (one injection of each) one hour prior to injection of Y^{91} results in the fixing in their skeletons of less than 10% of the radioelement fixed by the skeletons of the untreated controls in the same period.

Results of experiments in which EDTA treatment was initiated ½, 1, and 3 hours before Y^{91} injection are also presented in Fig. 2.**

The concentrations of Na-EDTA and Ca-EDTA present in the blood and soft tissues one hour after their administration is apparently sufficiently high to complex the injected Y^{91} soon after it reaches the blood stream. The yttrium-EDTA complex is then excreted from the body. As a result of this treatment, most of the injected yttrium is removed from the body before it can be fixed by the skeleton.

This is in contrast to Part 1 of these studies in which removal of bone-fixed Y^{91} was effected by the salts of EDTA.

3. CONCLUSIONS

The following conclusions may be drawn from the two parts of this study.

Removal. The possibility of effecting removal of substantial amounts of the fission product Y^{91} has been shown, using treatment with Na-EDTA followed by Ca-EDTA. Doubling the daily frequency of this treatment resulted in no significant increase in the removal of the isotope from bone.

Prevention. Administration of Na-EDTA, followed by Ca-EDTA starting one hour after Y^{91} injection, results, after 5 days, in a skeletal Y^{91} content in the treated animals which is only one-third that in the skeletons of the controls. The sooner the EDTA salts are administered following Y^{91} injection, the lower will be the final skeletal Y^{91} content.

Action of EDTA salts here is twofold: (1) By rapidly complexing the Y^{91} present in the blood and soft tissues, the EDTA greatly reduces (by excretion of the yttrium-EDTA complex) the Y^{91} available for deposition in the skeleton; (2) the EDTA brings about removal of part of the Y^{91} already fixed by the skeleton.

The advantage of administering the EDTA early is shown by its marked effect when given immediately (while a large part of the dose of the radioisotope is in the soft tissues and blood), as compared with its effect when administered 3 hours after injection when the Y^{91} is largely fixed by the skeleton.

Administration of Na-EDTA followed by Ca-EDTA one hour prior to the injection of Y^{91} results, 3 days after the Y^{91} injection, in an yttrium concentration in the skeleton of the pretreated animals of only 7% that found in the skeletons of the controls. Shortening the EDTA pretreatment interval from 1 to ½ hour prior to Y^{91} injection gives no enhanced effect in preventing the skeletal deposition of the injected yttrium. Significant lowering of the skeletal Y^{91} content was obtained also when pretreatment with single injections of both Na-EDTA and Ca-EDTA were given 3 hours before Y^{91} injection, although this effect was not as marked as that produced by pretreatment 1 hour before the yttrium injection.

The action of the salts of EDTA here is to complex rapidly the Y^{91} in the blood. This yttrium complex is excreted, and thus the yttrium is not available for deposition in bone.

Experiment D suggests the feasibility of a preventive approach, i.e., the use of prophylactic measures to prevent the deposition of fission products if exposure is contemplated, or even during prolonged exposure. In like manner, experiment C indicates that starting EDTA treatment within 1 hour after exposure to radioyttrium can effectively reduce the internal radiation hazard.

* * *

The opinions or assertions contained herein are the private ones of the authors and are not to be construed as official or reflecting the views of the Navy Department or the Naval Service at large.

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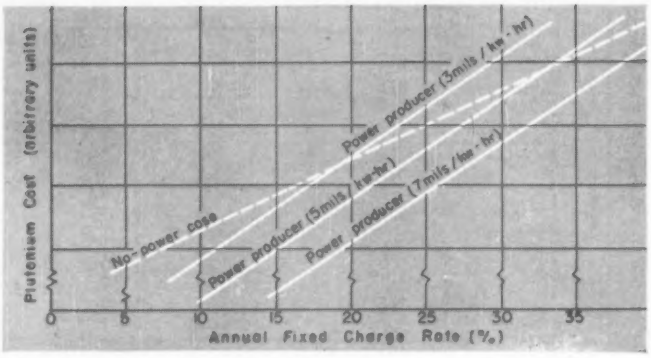
There is considerable debate today on the type of reactor that should be used for producing electric power. Here is one opinion on multipurpose reactors



Effect of fixed-charge rate on the plutonium cost

The Role of Multipurpose Reactors

By CHAUNCEY STARR
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USEFUL POWER generated through the nuclear fission process has always been one of the goals of the atomic energy field. The generation of such electrical power involves not only the problems of technical feasibility, but also economic and policy problems associated with the existence of competitive power sources. The value of nonmilitary power generated in such plants will be determined by competitive economic factors and the long-range utilization of national resources.

There is very little doubt at the present time as to the basic technical feasibility of generating central-station electrical power from atomic energy.* Although the practical engineering problems associated with such plants are many, novel, and frequently difficult, they are in the main susceptible to solution by the ingenious application of existing techniques and the results to be expected from well-defined development programs. However, to make central station atomic plants economically competitive with conventional plants, and to utilize our national resources fully, a considerable amount of bold engineering lies ahead.

It is of interest, therefore, to examine the constraints imposed upon this engineering field by the economic problems. Because there exists today no atomic power industry or any large body of unclassified information which would permit accurate projections into the future, such an examination must necessarily be approached with many reservations. Nevertheless, the subject is neither so complicated nor so vague as to prevent a judicious evaluation of future trends and probabilities.

Military Economy

When the military requirement of weapons material is the paramount objective of an atomic energy program, the nuclear reactors must be operated so that plutonium is produced at a minimum cost. It has always been obvious that if the energy associated with the operation of the reactor could be converted into electrical power, such multipurpose units might be a national asset. There is always a question under such circumstances of whether the necessary technological changes to produce by-product useful power will actually increase the cost of plutonium production to more than the return from the sale of power.

The economics of this situation involves not only the technical performance of such multipurpose plants, but also the nature of the financing associated with these plants. The graph on this page illustrates a typical situation for a multipurpose reactor which can be operated either for the production of plutonium only, or can be altered to produce useful electrical power. In this graph, plutonium costs on an arbitrary scale are plotted against the annual fixed-charge rate for the plant investment under various power sales values.

The slope of the curve is greater for the power-producing cases because the required investment is higher. The additional investment is required for the electrical generating equipment and for a more expensive reactor cooling system. As shown on the graph, this additional cost is offset by the sale of power if the fixed-charge rate is not too high. If the power sells for 5

* W. H. Zinn. Basic problems in central-station nuclear power, NUCLEONICS **10**, No. 9, 8 (1952).

mill/kw-hr, and the fixed-charge rate is about 32% per year, the plutonium from a power producer would cost the same as that from a reactor that produces no useful power.

It is clear from this illustration that, for very short plant life and for low power rates, the cheapest plutonium is produced without the by-product generation of useful power. Either increasing the plant life or increasing the sale price of power would make such a multipurpose reactor economically desirable.

It is of interest to compare on these curves the effect of various financing costs. The low fixed charges associated with government financing increase the desirability of a multipurpose reactor. Such multipurpose plants would probably be utilized as base-load plants in any network, since the primary incentive is the manufacture of plutonium, and a high load factor is therefore very desirable.

It has been suggested that due to the technological uncertainties and lack of experience in the operation of these plants, the nuclear reactor might at present have to be paralleled with a standby hydrocarbon-fuel steam plant. For such a system to be economic, the cost of nuclear power must be no greater than the fuel and operating cost in the conventional system, and under such a mode of operation the incentive for multipurpose operation is considerably reduced.

An additional limitation is placed on the multipurpose reactor by the maximum block of power permitted in the network involved. The larger the reactor, the lower are its unit costs, but as it is a single heat source it must be limited in size. For example, if an electrical block of 200,000 kw capacity is the maximum that can be handled, the reactor cannot be larger than about 600,000 kw of thermal power.

The allocation of costs between plutonium and power in a multipurpose reactor is to a great extent arbitrary, and with a fixed price of power in a network the major profit incentive arises from the opportunity to reduce the manufacturing cost of plutonium. This assumes, of course, that a demand for plutonium at a competitive price exists. In addition, the by-product chemical industry involving the radioactive fission products also offers an economic opportunity associated with reactor operation. This by-product industry cannot be economically evaluated at the present time.

Nonmilitary Economy

In a nonmilitary economy, power becomes the principal objective and plutonium manufacture has an industrial market only as an atomic fuel. For complete utilization of our resources, the very-long-term energy requirements would make it desirable to operate the breeder-type of plant. There is ample uranium-235 to start such breeder plants. If one assumes that an investment of several grams of U^{235} is required per kilowatt of reactor thermal-power capacity, it would appear that the U^{235} available in our estimated resources is sufficient to permit initiating the electrical generating capacity to handle the projected requirement 50 years hence.

In fact, the existence of atomic-power breeders with a breeding ratio of unity (just self-sustaining) would be sufficient to increase greatly the uranium resources available by raising the permissible price for natural uranium. Let us look at the price the breeder plant would be willing to pay the miner for expanded ore supplies. Assume, for example, that a cost of 0.3 mill/kw-hr was permissible for the feed material to a breeder plant. As 1 gram of completely utilized natural uranium produces 23,000 kw-hr of heat, or roughly 7,000 kw-hr of electricity, it would be permissible to pay $7{,}000 \times 3 \times 10^{-4} = \$2.10$ per gram. This is $950 per pound and is several magnitudes greater than present costs. Under such circumstances, the available natural-uranium resource of the United States would probably increase in the same proportion as the permissible costs, and would thus outstrip coal as an energy reserve. A similar case would undoubtedly exist for thorium.

The situation of the converter in the nonmilitary power economy is not unlike that of the self-sustaining breeder, although it is not quite as desirable. The production of new fuel (Pu or U^{233}) in the fertile material (U^{238} or Th) of a converter effectively multiplies the energy available from the U^{235} consumed. Let us, for example, assume a multiplication of 14. Then the cost per kilowatt-hour of the fuel fed to the converter plant would be a factor of 10 greater than that in the breeder, as only $\frac{1}{14}$ of the total supply is utilized. But if the price of the natural uranium or thorium is less than $95 per pound ($\frac{1}{10}$ of the price assumed above), the contribution to the cost of power still does not exceed the 0.3 mill/kw-hr used in the breeder example. Therefore, when the cost of natural uranium or thorium is as low as it undoubtedly is at present, the converter would produce power that may be quite competitive with that from the self-sustaining breeder.

The economically available uranium and thorium resources are therefore expandable by large factors as an approach is made to good neutron economy in converters or breeders. This dynamic relationship between our fissionable-material resources and the nuclear performance of power reactors is similar to the existent relationship between the availability of hydrocarbon fuels and the thermal efficiency of conventional plants.

The use of the converter to initiate the atomic power industry has many advantages:

1. The U^{238} not fully utilized by the converter is not wasted, since it may be stored and used as a fertile material for breeders at a later date—thus national resources are not destroyed.

2. Further, the converter is technologically less difficult than the breeder, since the problems of neutron economy are less demanding. For this reason, present reactor technology is generally adequate for atomic power plants which can be built now.

3. Finally, by their very nature, converters permit alteration of their operating cycles so as to produce primarily weapons material, with power as a by-product, from natural or low-enrichment uranium. That is, a converter may be designed to satisfy the requirements of both the military and the nonmilitary economy without any drain on the stockpile of weapons material. Although the value of such flexibility is a question of national policy, the present state of international affairs would appear to make it desirable.

Cost Comparison

It is of interest to consider a U^{233}-thorium breeding system with a breeding ratio of unity based upon present costs. A reasonably optimistic comparison of the typical costs which might be associated with the production of power in such a central-station plant, and a conventional steam plant,

is shown in the table on this page. These costs are primarily intended to establish the magnitudes involved and should not be considered definitive.

In this comparison, assumptions have been made concerning the present costs of the nuclear plant, involving a liquid-metal-cooled reactor and high-efficiency power-conversion equipment. A 20-yr life of all components of the plant, and typical private financing rates, have been assumed. It has also been assumed that about 2 gm of U^{235} is invested per thermal kilowatt of power, or about 6 gm per electrical kilowatt. The nominal cost of $20 per gram mentioned in the literature is used. The comparison indicates that the power from this type of nuclear power plant might almost be competitive with that from conventional plants.

It would appear from the comparison shown in the table that the capital investment required in a nuclear power plant of this type is almost double that required in a conventional steam plant. The operating costs, however, are significantly lower for the reactor plant because a large fraction of the fuel costs are eliminated. Engineering development will eventually reduce some of these costs. It is not unreasonable to assume that improvements in the cost of liquid-metal equipment and reactor construction can be achieved. In addition, the reactor fuel inventory includes the inventory associated with the fabrication of fuel elements and their chemical processing. Developments in processing technology will undoubtedly reduce this inventory considerably. However, it is not likely that significant reduction in operating costs of the nuclear plant will occur.

Although there does not appear to be a present need for breeders with breeding ratios much greater than unity, the existence of a market for fissionable material as a fuel provides an economic incentive to strive for high breeding gains. The market value of fuel produced by breeders would necessarily compete with that of fuels produced by isotope separation plants.

It is both stimulating and significant that, based upon what we consider a realistic analysis of the future possibilities of atomic energy, we can produce electrical power at a cost almost competitive with conventional present-day plants. In view of the inevitable increase in the cost of conventional fuels, as time goes on, there is a good likelihood that atomic power will play a major role in increasing power resources and reducing their total cost.

Comparison of Conventional and Atomic Power Costs

			Conventional plant			*Reactor power plant*		
	Life (yr)	*Fixed-charge rate (%)*	*Capital investment ($/kw)*	*Annual cost ($/kw-yr)*	*Power cost (mills/kw-hr)*	*Capital investment ($/kw)*	*Annual cost ($/kw-yr)*	*Power cost (mills/kw-hr)*
Capital Costs								
Reactor and auxiliary equipment	20	13.5*				50.00	6.75	0.96
Liquid-metal cooling circuit	20	13.5				60.00	8.10	1.16
Conventional boiler equipment	20	13.5	100.00	13.50	1.93	51.00	6.90	0.98
Electrical generating equipment	20	13.5	90.00	12.10	1.73	90.00	12.10	1.73
Coal handling equipment	20	13.5	10.50	1.40	0.20			
Land and land improvement	20	13.5	10.50	1.40	0.20	10.50	1.40	0.20
Reactor fuel inventory	∞	10.5				120.00	12.60	1.80
Total investment costs			211.00	28.40	4.06	381.50	47.85	6.83
Operating Costs								
Fuel (coal at $7.18 per ton)				20.20	2.88			
Reactor fuel processing†							2.50	0.36
Reactor operation and maintenance							1.65	0.24
Power plant operation and maintenance				3.30	0.47		3.30	0.47
Total operating costs				23.50	3.35		7.45	1.07
Total power costs					7.41			7.90

* This includes 3% for depreciation, based on a 5% sinking fund, 6% for profit, and 4.5% for taxes and insurance.

† Includes recovery of fuel and waste disposal.

Summary

The present atomic power picture may roughly be summarized as follows:

1. The generation of electrical power through nuclear power plants is completely feasible at the present time.

2. The fissionable material resources of the United States may be as large as our coal reserves in energy content and could play a significant role in the power requirements of the country for the next several hundred years. In addition, these economically recoverable resources may expand many times when the self-sustaining breeder contributes to our power production.

3. On the assumption that the special value of plutonium for military purposes will continue for at least 10 years, the multipurpose plan of using converters is the most promising approach to initiating an atomic power industry. It would contribute not only to the power resources of the country, but also to its military strength through the production of low-cost plutonium. In addition, such plants would always be standby sources for the manufacture of atomic weapons material if such is required in the future. However, from the point of view of a long-range atomic power industry, it would appear undesirable to depend upon the military requirements for plutonium for the economic justification of a nuclear power plant.

4. As indicated above, the limitations on natural resources of uranium would indicate that a nonmilitary atomic power industry should be based at present upon converters, and then, when development permits, upon breeders with breeding ratios of unity or more. Both types of plants could establish a permanent base for economic atomic power in the near future. To reduce the cost of power to a minimum and eventually to assist in the expansion of the atomic power industry, development of breeders with high breeding ratios should be undertaken.

5. Finally, to get the maximum long-term usage of our fissionable material resources, breeders utilizing both uranium and thorium should be built.

EQUIPMENT FOR NUCLEAR PHYSICS

EKCO SCALER, TYPE N526

This is a new general-purpose scaling unit, comprising two high-speed electronic decades followed by a resettable electro-mechanical register and incorporating pulse height discrimination and paralysis facilities.

The overall counting speed of the scaler is limited by the mechanical register to about 1000 per second but can be improved by the use of an external high-speed register. Alternatively, scalers can be connected in series when evenly spaced pulses are counted up to 600,000 per second.

The new Ekco Scaler Type N526 is developed from the 1009 unit which it replaces.

SPECIFICATION

Sensitivity. The minimum pulse amplitude is 5 volts positive. The minimum operating pulse width is 0.25 microseconds with a minimum interval of 1.5 microseconds between pulses.

Discrimination. The discriminator permits only those pulses whose amplitude exceeds the pre-set level to be counted. The level can be set at any voltage from 5 to 50.

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Nomogram for the Critical Equation

By F. T. MILES* and HARRY SOODAK†

In the Fermi age theory for thermal piles the criticality condition is expressed by‡

$$ke^{\tau\Delta} - 1 + L^2\Delta = 0 \qquad (1)$$

In this equation k is the multiplication constant for an infinite pile, τ is the age, and L is the diffusion length. These quantities are functions of the materials used and are calculable, in principle, from nuclear constants and lattice geometry.

Δ is the critical Laplacian or buckling, and its value determines the critical size of the pile.

Since Eq. 1 is transcendental, the solution for Δ is generally obtained by successive approximations. This may be time consuming, particularly if a range of pile designs is being considered. The nomogram at the left below gives solutions of Eq. 1 directly. A straight edge through the values of k on the left-hand scale and of L^2/τ on the right-hand scale intersects the appropriate scale at the value of $-\tau\Delta$ which satisfies Eq. 1. To expand the scales in a limited space, three scales for different ranges of k have been superimposed and separately numbered. Each is used with the corresponding scale for $-\tau\Delta$.

Example: Find the critical size of a bare spherical pile from the following data as fixed by the materials: $k = 1.348$, $L = 12$ cm, $\tau = 200$ cm^2.

Connect 1.348 on the k_B scale and $(12)^2/200 = 0.72$ on the L^2/τ scale, with a straight edge. This intersects the $-\tau\Delta_B$ scale at 0.178_0. Therefore

$$-\Delta = \frac{0.178_0}{200} = 8.9_0 \times 10^{-4}\ \text{cm}^{-2}$$

The radius of the critical sphere is then‡

$$R = \frac{\pi}{\sqrt{-\Delta}} = \frac{3.1416}{(8.9_0 \times 10^{-4})^{1/2}} = 105._2\ \text{cm} \qquad (2)$$

Where k is close to 1 the nomogram cannot be read with high percentage accuracy, but in this range (and for $L^2/\tau > 2$) the approximate solutions

$$-\Delta = \frac{k-1}{L^2+\tau} \qquad (3)$$

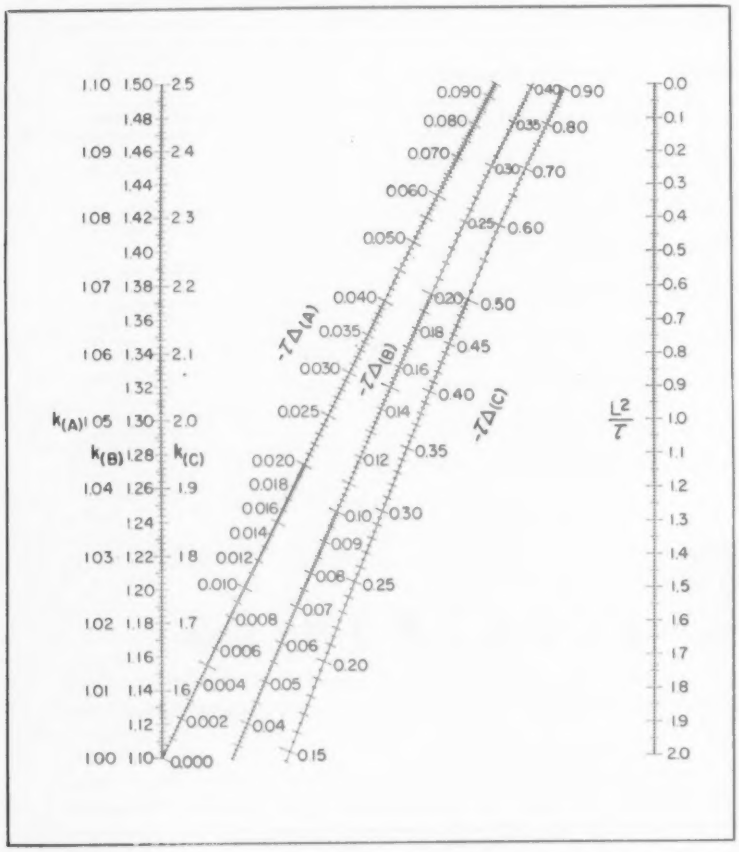
and

$$-\Delta = \frac{k-1}{L^2+k\tau} \qquad (4)$$

are sufficiently precise.

* * *

Large-scale prints of the nomogram are available from NUCLEONICS *on request.*



NOMOGRAM for the critical equation $ke^{\tau\Delta} - 1 + L^2\Delta = 0$. Straight edge through given k and L^2/τ intersects proper curve to give resulting value of $-\tau\Delta$

* Brookhaven National Laboratory, Upton, N. Y.

† Physics Department, College of the City of New York, New York.

‡ H. Soodak, E. C. Campbell, "Elementary Pile Theory," p. 42 *et seq.* (John Wiley & Sons, Inc., New York, 1950).

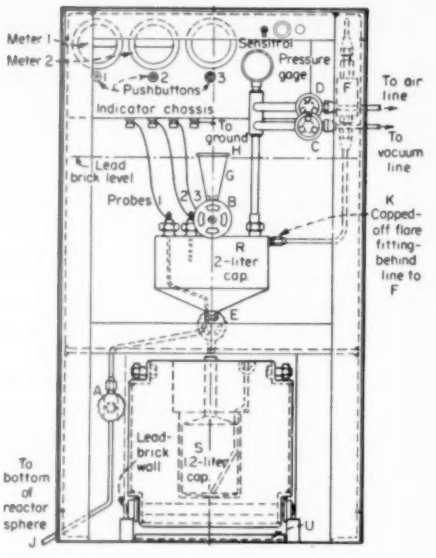


FIG. 1. Front view of solution handling system with movable lead cart in place

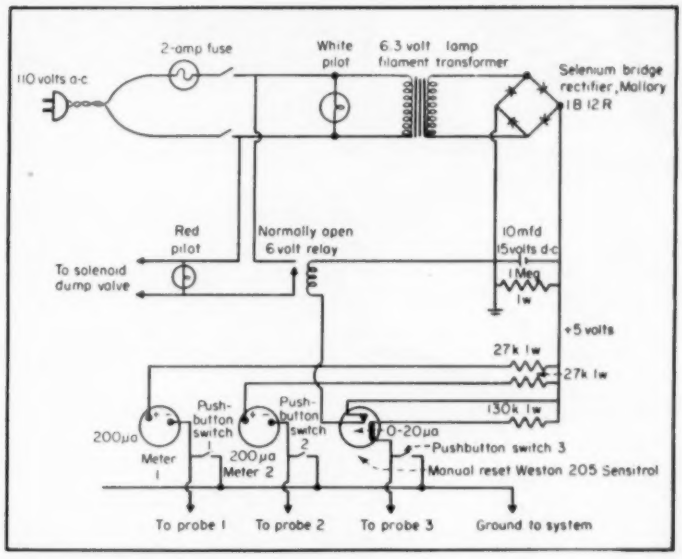


FIG. 2. Schematic diagram of electrical fixed level indicators and safety arrangement

Solution Handling System for a Homogeneous Reactor

By J. E. EVANS, L. D. P. KING, and J. W. SCHULTE
Los Alamos Scientific Laboratory, Los Alamos, New Mexico

For use with the Los Alamos Homogeneous Reactor (Water Boiler), a safe and convenient system for handling the radioactive solution of the reactor was essential. Such a system is needed for:

1. Adding or withdrawing solution when the reactor is being brought up to critical for the first time.
2. Withdrawing small samples for chemical analysis.
3. Adding nitric acid to keep the uranyl nitrate from precipitating out of the solution.
4. Adding or withdrawing extra uranyl nitrate to increase or decrease the reactivity of the reactor for special experiments, or for other reasons.
5. Withdrawing the entire amount of solution for major reactor repairs or for reprocessing the solution.
6. Returning the radioactive or new solution to the reactor.

In the past, haywire systems have resulted in several spills of radioactive solution. This article describes the stainless-steel system and techniques now being used. After a year of use, this system has resulted in no spills and no cases of excessive radiation exposures to personnel.

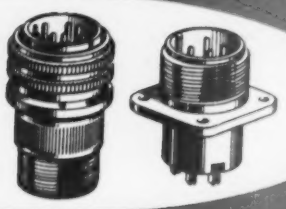
General Principles

The general idea of the system is simple (see Fig. 1). A small tube *J* from the bottom of the reactor sphere leads up through the reactor shielding and connects into the bottom of a 2-liter reservoir *R* located on the top of the reactor. The pressure in *R* can be reduced by means of a vacuum pump to draw solution into it from the sphere. A pipette can be inserted through gate valve *B* for the withdrawal of small samples ($\leq$2 ml). Larger samples are allowed to flow through *E* into the shielded container *S*.

The pressure in *R* can also be increased by opening the valve *D* to the air line to force the contents of *R* and connecting tube *J* into the sphere. By alternately applying suction or air pressure to *R* and *J*, we have a method for thoroughly mixing new additions to the solution or withdrawing representative samples. The entire system is shielded by about 4 to 6 in. of lead.

A more detailed explanation of the system can be given by a discussion of the procedure used in withdrawing small samples for chemical analysis. Before beginning any operation involving the solution, the indicator chassis (Fig. 1) is turned on and checked. Push button *1* is closed to see that indicator meter *1* will read when solution comes in contact with probe *1*. Meters *2* and *3* are also checked. The electrical circuit used is shown in Fig. 2.

It is first necessary to get rid of any radioactive gas which has collected just below valve *A*. This gas is caused by fissions which occur in that portion of the solution which is in the tube *J* leading from the bottom of the sphere to valve *A*. The gas is flushed out by forcing air down through the tube and bubbling it through the solution in the sphere. To do this, first valve *D* is opened; this connects the 2-liter reservoir *R* to the filtered air line through a reducing valve set to deliver a maximum pressure of approximately 3 psi. Now, when valve *A* is opened, the con-



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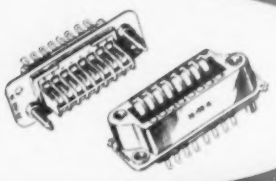

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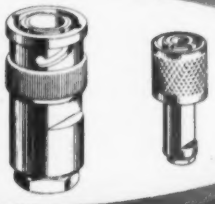

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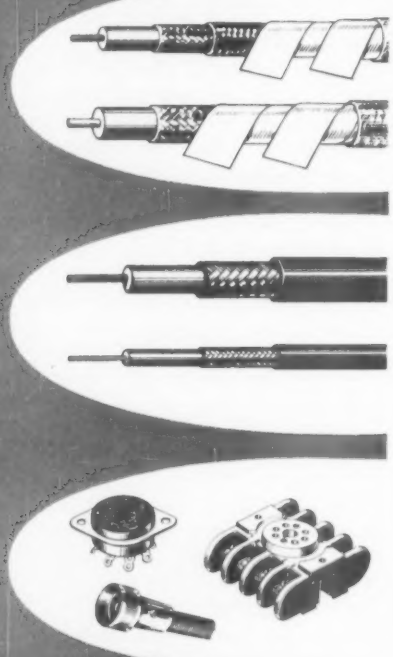

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tents of the stainless-steel tube below it are forced back into the sphere. The gaseous radioactive fission products which have collected in the tube below *A* are bubbled through the solution and into the gas circulating system.

Next, valve *D* is closed and valve *C* is opened to reduce the pressure in *R* enough to allow some of the sphere solution to flow into *R*. When the solution level in *R* reaches probe *1* the first 200-microampere meter will read about full scale. Valves *A* and *C* are then closed, holding about 10 ml of solution in *R*.

This sample is rejected because it is probably not representative of the sphere solution. There may be droplets of solution or of rinse water on the surfaces of the system left from a previous operation. These droplets would change the composition of the sample. Therefore, this first liquid sample is forced back into the sphere and a new sample is drawn into *R*. Any droplets on the surfaces of the system when this new sample is drawn up should have nearly the same composition as the sphere solution. Repetition of this operation two or more times should give a representative sample.

The gate valve *B* can then be opened, the pipette inserted into the stainless-steel cone *G*, and a small amount of solution drawn up into the pipette by means of a hypodermic syringe. Figure 3 shows a cross section of the shielded pipette now being used. After the closing of valve *1* of the pipette, valve *A* is opened to allow the solution in *R* to flow back into the sphere and away from the tip of the pipette. This process can be accelerated by opening valve *D* a little. Next, a little suction is applied to the pipette and valve *1* is opened to draw the solution out of the tip up into the region between valves *1* and *2* of the pipette. After the closing of valves *1* and *2*, the pipette is withdrawn and the glass tube is placed over the tip. Then it can be carried away to the chemical laboratory for analysis.

The specially built rinser is now put in through the cone *G*, and 5 ml of H_2O are sprayed over the walls of *R*. The rinser is shown in Fig. 4. The purpose of this rinse is to keep the reservoir clean; otherwise, nitric acid fumes form conducting films over the Teflon insulators at the probe inlets. Also, uranium deposits on the wall would make the next sample withdrawn non-representative of the sphere solution. After the wash water is bubbled

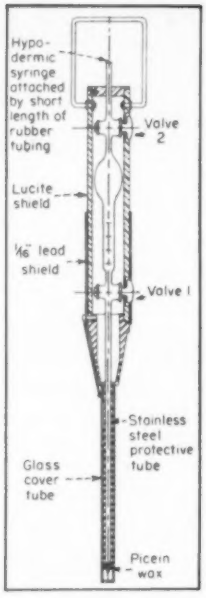

FIG. 3. Shielded pipette

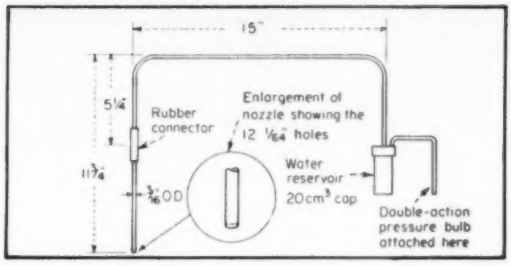

FIG. 4. Stainless-steel rinser

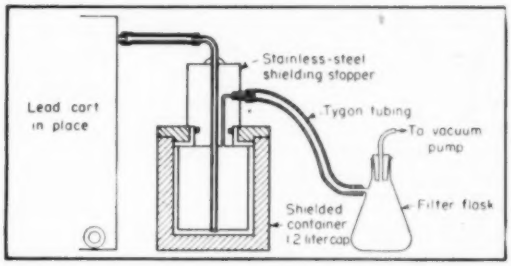

FIG. 5. Shielded container and arrangement for removing 1-liter lots of solution

through and mixed with the sphere solution all valves are closed, the dirt cover *H* is replaced, and the electrical indicator chassis and the vacuum pump are shut off.

The probe insulators have been designed to withstand pressures of at least 2 atmospheres. The entire system which comes in contact with the sphere solution was made from type-347 stainless steel, and all stainless parts were pickled in concentrated HNO_3 after all machining and welding had been completed. The assembled system was thoroughly leak-tested before being placed in use.

The combination vacuum and pressure gauge is a brass unit. It comes in contact with some dilute nitric acid fumes but it has worked satisfactorily over a 1-year period. The electrical indicator and safety chassis have been designed so that 5 volts is the maximum that can be applied to the probes. All bare contacts and terminals have been coated with glyptal so that even a dangling broken wire carrying 110 volts inside the chassis could not put 110 volts on either of the probes. The purpose of the low voltage is to minimize the chance of producing a spark inside *R*. *R* might conceivably contain an explosive gas mixture if valve *A* were left open through several hours of operation.

The three probes in *R* are set at levels of about 10 ml, 1 liter, and 1.2 liters. Probe *3* is part of a safety feature. If 1.2 liters of solution are accidentally brought up into *R*, the contacts of the Weston Sensitrol are closed. This puts 5 volts d-c on the relay coil, which in turn puts 110 volts a-c on the solenoid valve *F*. This connects *R* to the atmosphere and returns the solution in *R* to the sphere.

The shielded pipette shown in Fig. 3 has proved adequate for solution samples up to 2 ml. The glass tube held over the pipette tip by means of the three rubber bands is for the purpose of catching any droplets of solution which might fall from the pipette tip during transportation from one place to another. The space between the glass pipette tip and its surrounding stainless-steel protective tube is sealed off at the tip by a wax seal. Paraffin and beeswax are not very satisfactory since they absorb quite a bit of the radioactive solution. Picein has been found to be much better.

Addition of Solution

The addition of solution that is not radioactive is quite simple. Gate valve *B* is opened and a funnel is put through *B* down into *R*. A stainless-steel funnel is more desirable than one made from glass, since there is always a possibility that someone will attempt to close the gate valve *B* without removing the funnel.

A funnel must be used for this operation to prevent solution or wash-water droplets from staying on the cone and tube walls in the vicinity of *B*.

After the solution has been poured into *R*, the funnel is removed and the gate valve *B* closed. Then *A* is opened and the liquid forced down into the sphere by a small pressure (3 psi) admitted through *D*. If the solution being added is anything other than water, it should be washed from the walls of *R* with the rinser as described previously.

Withdrawal and Storage

The withdrawal of large amounts of the radioactive solution poses some additional problems. The gamma activity of the solution is about 20 mc/cm³ about 40 hr after shutdown. This figure is for operation at a power level of 25 kw.

The solution is drawn up into *R* until it reaches probe *2*. Then valve *A* is shut off. Next, valve *E* is opened, and the solution is drained into a stainless-steel can *S* which is in a heavy lead cart.

It was originally intended that for solution removals in excess of 1 liter, the shield cart would be transported back and forth after its contents had been transferred to other containers in a shielded storage vault. It is believed, however, that the over-all handling for large amounts of the solution will be simplified by having about 15 shielded stainless-steel containers of 1-liter capacity available. The solution will be transferred from the can in the large lead cart to one of these shielded containers. This will be accomplished by means of the setup shown in Fig. 5. The large lead cart need not be removed from its place, and the portable container and the tubing which carries radioactive solution can be adequately shielded by a temporary shield made of lead bricks.

Return of Radioactive Solution

The return of the solution to the reactor is a somewhat simpler process than that of removing the solution. A liter of solution in one of the shielded containers is placed beside the "removal system" on the top of the reactor shield. A tube is connected to the normally closed-off flare fitting *K*. The free end of the tube is put into the shielded container so that it reaches the bottom of the receptacle. By reducing the pressure in reservoir *R*, the liter of solution is transferred into the reservoir and then lowered into the sphere.

* * *

We would like to acknowledge the assistance of J. A. Bridge in the design and construction of this equipment.

BOOKS

Practical Radiography for Industry

By H. R. CLAUSER (Reinhold Publishing Corporation, New York, 1952, x + 301 pages, $7.50). Reviewed by ADAIR MORRISON, Arthur D. Little, Inc., Cambridge, Mass.

After extensive application during the war, followed by a period of much reduced activity, the use of industrial radiography is again expanding and many new technicians must be trained. For such training, this book is timely and it is a suitable introduction to the physical principles and practical techniques of industrial radiography.

The author has done a skillful job of assembling and arranging the information. The first half of the book is a clear exposition of the physical properties of X-rays, the factors entering into the radiographic exposure and affecting the quality of the radiograph and the sensitivity obtained in it, the characteristics of films and intensifying screens, and the problems of processing and viewing the radiographs. The second half deals with specific applications of X-rays and gamma rays and includes chapters on the examination of castings and welds, radiography with gamma rays and with high-voltage X-rays, fluoroscopy, and other techniques and applications. A final chapter discusses protection and safety.

The text is well arranged and clearly written, and there are many illustrations. A considerable number of radiographs, notoriously difficult to reproduce on paper, have been quite successfully printed, and these add greatly to the value of the discussion of particular types of defects.

The treatment of the various topics is well balanced and reasonably complete. Much specific information, the kind needed by the novice, is included.

Because of changes in the last two years, the chapter on gamma-ray radiography is somewhat out of date. Although there is a brief section on cobalt-60 as a substitute for radium, and mention of iridium-192 as a likely gamma-ray source, the discussion is largely limited to radiography with radium sources of less than one curie activity. The availability of multicurie sources of cobalt-60 and iridium-192 has changed the economics and the scope of gamma-ray radiography.

Cobalt-60 sources differ from radium in two respects: they are much less costly, and they are very much smaller in volume, for a given radiation intensity. Sources with activities of 10 to 20 curies per gram, and higher, are now available; per unit volume, they have radiation outputs 50 to 100, or more, times greater than has radium. Thus, for sources of equal physical size, exposure times may be correspondingly reduced. Because of the lower cost, sources of 10 to 100 curies of activity are practicable. The short exposure times possible with such sources make gamma rays much more competitive with X-rays. Such sources present new problems in protection, safety, handling and storage.

Iridium-192, in wide use in Britain by 1950 and now available here, extends the range of gamma-ray radiography to sections much thinner than can be examined with radium or cobalt-60; in fact 2% sensitivity can be obtained in steel down to 0.1 inch of thickness. Radiation intensities for a given geometrical size are higher even than can be secured from cobalt-60.

Other isotopes, and fission products, with their special properties and advantages, are expected to extend still further the usefulness of gamma-ray radiography.

Thus, this chapter will require considerable revision in a second edition

The permissible exposure or dose now accepted by the AEC and radiologists is 50 milliroentgens per day, superseding the figure of 100 mr/day used by the author. This will require some modifications in his chapter on protection and safety.

References are given at the end of most chapters; they could well be supplemented by an appendix containing a selected and annotated bibliography of books and articles to which the reader could turn for additional information. Such a bibliography might mention other books similar in purpose to this, such as Crowther's "Handbook of Industrial Radiography" and the books by St. John and Isenburger, by Low, and by Zmeskal.

BOOKS RECEIVED

Nuclear Stability Rules, by N. Feather, Cambridge University Press, New York, 1952, ix + 162 pages, $4. (*To be reviewed.*)

Angewandte Radioaktivität, by K. E. Zimen, Springer-Verlag, Berlin, 1952, viii + 124 pages, DM 18.80.

This is the newest German-language text on the fundamentals of radioactivity and its application in medicine, biology, chemistry, physics, and industry. A pull-out chart of the isotopes and other tabulations of useful data from references dated through 1951 are included. Otto Hahn has written an introduction for the text.

Thermionic Vacuum Tubes and Their Applications, 6th ed., by W. H. Aldous and Sir Edward Appleton, John Wiley & Sons, Inc., New York (Methuen & Co., Ltd., London), 1952, vii + 160 pages, $2.

A completely revised and rewritten edition of a monograph first appearing in 1932, directed to physics students who have not made a special study of radio-frequency phenomena.

Theory of Electric Polarisation, by C. J. F. Böttcher, Elsevier Publishing Company, Houston, Texas, 1952, xiii + 492 pages, $10.

The author, a professor of physical chemistry at the University of Leyden, The Netherlands, has directed his text to the theoretician, researcher, and advanced student. To serve both chemist and physicist, he restricts discussion to the "classical" theory of dielectrics but manages to emphasize the connection between theory and experiment. Four appendixes deal with associated mathematics.

Essentials of Microwaves, by Robert B. Muchmore, John Wiley & Sons, Inc., New York, 1952, vi + 236 pages, $4.50.

A well-illustrated survey of microwave apparatus and its functions is offered in this general-purpose text that presents the physical picture with a minimum of mathematics. It is directed to a broad audience, with applications in many fields being discussed; examples: linear accelerators, spectrometers, magnetrons, radar, etc.

Superconductivity, 2d ed., by D. Shoenberg, Cambridge University Press, New York, 1952, x + 256 pages, $6.

The many advances in our understanding of superconductivity since the first edition of this tract was written in 1938 made it necessary to issue a complete revision of the text for this, the second edition. The magnetic properties of macroscopic superconductors, thermodynamic and other

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*Max. Perm.	–	100	183	1100	1030	1710	3300	4300	3800	3000	750
*Sat. Flux Density	Gauss	1500	1900	4200	3100	3800	3200	3400	2800	2000	2900
*Residual Mag.	Gauss	1000	830	2700	1320	1950	1050	1470	1500	700	1600
*Coercive force	Oersted	5.0	3.0	2.1	1.0	.65	.25	.18	.35	.30	.80
Temp. Coef. of initial perm.	%/°C	.15	.04	.40	.30	.25	1.3	.66	.80	.30	.22
Curie Point	/°C	300	260	330	165	160	160	150	125	70	180
Vol. Resistivity	ohm-cm.	$1x10^7$	$2x10^5$	$2x10^4$	$3x10^7$	$4x10^5$	$1.5x10^8$	$1x10^4$	$2x10^4$	$2x10^5$	$5x10^7$
Loss Factor:											
At 1 mcs/sec	–	.0005	.00016	.00007	.00005	.00008	.00008	.00030	.0004	.0003	.000055
At 5 mcs/sec	–	.0007	.0011	.0008	.0012	.002	.00075	.00155	.001	.005	.0004

*Measurements made on D.C. Ballistic Galvanometer with Hmax 25 oersteds. Above data is based on nominal values.

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thermal properties, structure of the intermediate state, depth of penetration of a magnetic field into a superconductor, and theoretical aspects are the major topics discussed. Many references to recent work are given.

A "First Book" on Fire Safety in the Atomic Age, by Horatio Bond, National Fire Protection Association, Boston 10, Mass., 1952, viii + 72 pages, $3.

The heat and secondary effect of the blast from atomic and hydrogen bombs have created new problems of safety, which are here set down for the fire protection engineers in industry and on the home front.

Thermal Diffusion in Gases, by K. E. Grew and T. L. Ibbs, Cambridge University Press, New York, 1952, xi + 143 pages, $4.50.

Based closely on the theory of Chapman and Cowling, experimental methods and results are discussed in connection with the application of thermal diffusion to the separation of gas mixtures. Thermal diffusion in liquids and the diffusion thermoeffect are also considered. There are five appendices and four pages of references.

Statistical Thermodynamics, 2d ed., by Erwin Schrödinger, Cambridge University Press, New York, 1952, 95 pages, $1.75.

This is the text of lectures delivered by the author in 1944 at the Dublin Institute for Advanced Studies. Except for the addition of an appendix, this edition is the same as the first. The appendix serves to offer the general proof, according to Schrödinger, that the "thermodynamical functions depend on the quantum-mechanical level-scheme, not on the gratuitous allegation that" the sharp energy values ascribed to every system are the only states allowed.

ALSO OF NOTE

Atomic Power and Private Enterprise, Joint Committee on Atomic Energy print, 82d Congress, 2d Session, Dec., 1952. A comprehensive, 415-page compilation of material on atomic power problems, compiled by the staff of and for the use of the Joint Committee on Atomic Energy of Congress. All aspects of the problem are covered in excerpts from testimony, speeches, articles, etc., by administrative and technical specialists in the field. *GPO, Washington 25, D. C.*

An International Bibliography on Atomic Energy, Vol. 2, Scientific Aspects, Supplement No. 1. This publication of 328 pages lists over 8,000 references published in 1949 and 1950 and supplements Vol. 2, which listed some 24,000 articles and books on the scientific aspects of atomic energy published throughout the world between 1925 and 1949. *Columbia University Press, New York 27, N. Y., $3.50.*

Lecture Notes on Pile Theory, by C. A. Rennie (AERE R/L 5). A 36-page paper-bound report dealing with the basic nuclear design features of natural-uranium, graphite-moderated thermal reactors, published by the Atomic Energy Research Establishment, Harwell, England. *H. M. Stationery Office, 423 Oxford St., London, W. 1.*

Uranium, Plutonium, and Industry. This illustrated, 47-page pamphlet sponsored by the ASME's Nuclear Energy Applications Committee, is a good, nontechnical summary of the atomic-energy program in the United States. *The American Society of Mechanical Engineers, 29 West 39th St., New York 18, N. Y., $1.50.*

Atomic Energy Commission Document Price List No. 21 (issued October, 1952). A listing of all AEC research reports currently available for purchase. *Copy of price list free on request from Office of Technical Services, Dept. of Commerce, Washington 25, D. C.*

Low Temperature Physics (National Bureau of Standards Circular 519). Publication of 291 pages comprising proceedings of NBS Symposium on Low-Temperature Physics, March 27–29, 1951. Superconductivity, calorimetry, thermometry, second sound, and metals, methods, and instrumentation for low-temperature work are among the subjects discussed in the 61 papers recorded. *GPO, Washington 25, D. C., $1.75.*

Infrared Isotope Analysis, by J. C. Kluyver. It is suggested in this 86-page Dutch-English monograph that the difference in the infrared absorption spectra of isotopic molecules might offer a method for measuring abundance ratios of stable isotopes. An instrument built for the purpose at the University of Utrecht in The Netherlands is described, together with experiments made with it to determine the C^{13}-C^{12} ratio in carbon dioxide. *Drukkerij v/h Kemink en Zoon N.V., Domplein 2, Utrecht, The Netherlands.*

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NUCLEONIC EVENTS

Strike at Nickel-Tubing Plant Delays AEC Expansion; Forces Release of Classified Data

A three-month strike against the sole producers of nickel-plated process piping is causing considerable delay in the construction of Atomic Energy Commission facilities. The strike of 1,600 production, maintenance, and office workers at the American Locomotive Co.'s plant at Dunkirk, N. Y., has had such a serious effect on atomic energy construction that, under instructions from President Truman, the Department of Justice has obtained an 80-day strike injunction under the Taft-Hartley Act.

Because of the seriousness of the strike, AEC was forced to disclose for the first time the end-use of the products and materials turned out at the Dunkirk plant.

The President acted on the basis of the report of a board of inquiry which asserted that:

- Lack of the special process piping from the Dunkirk plant delayed construction of the C-33 and C-35 gaseous-diffusion plants at Paducah, Ky., and the K-33 plant at Oak Ridge, Tenn. It would take seven months to produce the internally-plated piping if manufacture were started now in an alternate plant.
- The strike also halted fabrication of heat exchangers and tube bundles needed to construct the Savannah River heavy-water plant. Even if production is resumed immediately, AEC estimates completion of one-third of the Savannah River project would be "materially delayed." Heat exchangers are used to control the temperature of the process streams between steps in the heavy-water production process.
- The shortage of replacement parts for heat exchangers and tube bundles has reduced output at the heavy-water plant at Dana, Ind.
- The strike halted production of gas converter subassemblies. This has delayed the assembly of converters at the Oak Ridge gaseous-diffusion plant and, if continued, will delay construction of the new Paducah and Oak Ridge plants. It would take a year to start production of gas converters at some other place.
- The Dunkirk plant also produces evaporators, coolers, condensers, and other fittings and devices used exclusively in construction of atomic energy installations.

Exactly how much of the atomic energy program was delayed by the strike is secret. The board of inquiry was convinced, however, that the strike affected "a substantial part of atomic weapon production," and of the production of industries making equipment and fissionable materials essential to the atomic energy program.

According to AEC, "the total estimated cost for the facilities whose construction is faced with delay by the Dunkirk strike exceeds the total cost of all existing facilities, under the jurisdiction of the United States, presently capable of producing large quantities of fissionable material."

Joint Congressional Committee Holds Series of Meetings on Atomic Energy; Discuss Atomic Power at Length

The questions of what kind of atomic power plants should be built and who should develop them are among those which have been under discussion in a series of confidential meetings that the Joint Congressional Committee on Atomic Energy has been having with electric utility and other industry groups, scientists, lawyers, and government officials.

As the first public announcement of the meetings, the Joint Committee has put out a report called "Atomic Power and Private Enterprise." The committee did not identify specifically what companies took part in the meetings, but the report makes clear that these included representatives of the five industrial teams now engaged in power feasibility studies (NU, June '51, p. 42; Oct. '52, p. 84), interested electrical equipment manufacturers, and makers of other products for the atomic energy program. Government representatives included spokesmen for the Atomic Energy Commission and the armed services.

On the question of the type of atomic power plant that should be built, there were two main points of view. One group considered any plant capable of driving a generator as a successful atomic power plant. The other group felt that a successful plant should produce electric power at a cost competitive with conventionally-fueled power plants—nothing less than 150,000 kilowatts at a cost of 5–8 mills per kilowatt-hour.

Some individuals, including equipment and utility company officials, were of the opinion that AEC should do the developmental work on the plants. They contend that AEC should assume responsibility for carrying out the development through pilot plant, or prototype plant, stages.

This brought up the question of whether or not AEC should undertake development with its own personnel. If it doesn't, it is said that the selection of contractors would raise the problem of giving unfair advantage to the contracting firms chosen to do the development work.

Utility men were divided as to who should bear the developmental costs. One group was willing to shoulder costs up to $10-million to $15-million, but considered expenditure of $100-million as unreasonable. Two other groups felt that the job is beyond the ability of any single company. They are trying to develop new company groups which can take on the developmental job.

The conferences clearly were another in a series of steps preparatory to full-dress hearings by the Joint Committee early this year. The first of these moves came last summer when AEC was advised that it was time to review the McMahon Act of 1946 and its administration by AEC. The recent conferences indicate that the forthcoming hearings will center on atomic power possibilities and private industry's role in that field.

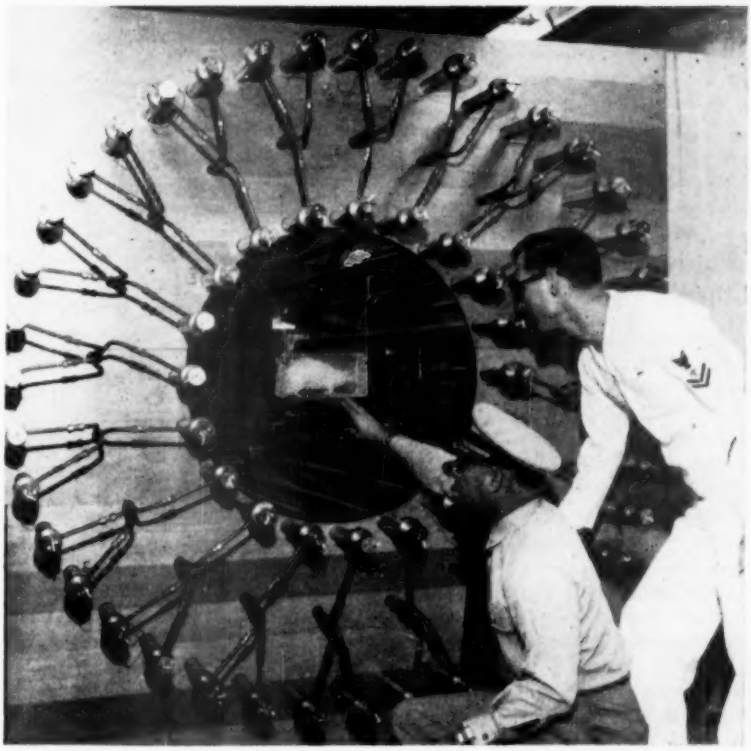
Research Grants Totaling $728,250 Awarded

The National Science Foundation has announced approval of 52 grants and two contracts, totaling $728,250, in the biological and physical sciences and to support studies and conferences on science and scientific education.

This is the first group of awards to be made under the foundation's budget for fiscal 1953. Additional proposals are being evaluated.

Grants and contracts in this group were made to institutions in 21 states and the District of Columbia. The duration of the research grants range from 6 months to 3 years, with an average length of 1.9 years. The average grant amounts to $6,300 per year.

The fields included in the list are astronomy, chemistry, developmental biology, earth sciences, engineering, genetic biology, mathematics, microbiology, molecular biology, physics, regulatory biology, systematic biology, research education in science, scientific information, and special studies in science.



GAMMA-RAY GENERATOR at Naval Medical Research Institute, Bethesda, Md., is designed to give reproducible uniform total body irradiation. The 60 tubes contain a total of 100 12-curie radiocobalt capsules. Tubes, which are of aluminum alloy, are 17 ft long, and capsules can be arranged in them in a spherical pattern surrounding the target. Capsules are loaded by remote control into ends of tubes behind 5-ft radiation shield and are positioned in the tubes by air pressure

Committee of Senior Reviewers Reorganized

The first thoroughgoing reorganization of the Atomic Energy Commission's committee of Senior Responsible Reviewers has just about been completed. Involved in the changes are specific terms of service for the committee members, reassignment of responsibilities, and the setting up of a group of subcommittees.

The committee, which advises AEC on classification and declassification of technical information, had its first shift of members last summer when two of the four original members resigned (NU, Sept. '52, p. 68). At that time, W. F. Libby, chemist, and R. L. Thornton, physicist, were replaced by J. R. Richardson, physicist, R. H. Crist, physical chemist, T. B. Drew, chemical engineer, and J. P. Howe, chemist. W. C. Johnson, chemist, became chairman of the committee. The other member is J. M. B. Kellogg, physicist.

Before the reorganization, the major responsibilities of the committee members were: plutonium project, Johnson; electromagnetic project, Thornton; military utilization, Kellogg; and gaseous-diffusion project, Libby.

This division of activities has been replaced with a more modern alignment covering seven major categories. The new assignments are: chemistry and chemical engineering, Johnson and Drew; physics, Kellogg and Richardson; metallurgy and solid state, Howe; reactors, Drew and Howe; isotope separation processes, Crist; raw and feed materials, Johnson; and weapons, Kellogg.

These assignments, like the old ones, do not limit the areas of activities of any of the members. All members will continue to share in decisions on major recommendations in all areas.

A group of subcommittees is being set up along the same lines as the division of responsibility of the committee members. The first of these subcommittees to be organized is that for reactors, and four members have already been appointed. They are: Paul Gast, Hanford Operations Office; John A. Swartout, Oak Ridge National Laboratory; Samuel Untermeyer, Argonne National Laboratory; and Thomas Snyder, Knolls Atomic Power Laboratory.

According to classification officers of AEC, additional members of the committee of Senior Responsible Reviewers, as well as additional subcommittees will speed the work of the panel.

Jerome D. Luntz Named Editor of "NUCLEONICS"

Jerome D. Luntz, formerly executive editor of NUCLEONICS, succeeded Keith Henney as editor on January 1. Mr. Henney is now editorial director of both NUCLEONICS and *Electronics*.

Mr. Luntz joined the staff of NUCLEONICS in July, 1947, as assistant editor, was named managing editor in January, 1950, and has been executive editor since June of that year. He holds an engineering degree from the City College of New York, and worked

for Ford, Bacon & Davis at Oak Ridge during World War II. The last two years of the war he was engaged in instrument research and development for the Cleveland research laboratory of the National Advisory Committee for Aeronautics. Immediately prior to joining NUCLEONICS, he taught electrical engineering at the University of Minnesota.

Uranium Rods Burst in Chalk River NRX Reactor

A burst in some uranium rods in the NRX reactor at Chalk River, Canada, last month brought forth the first public announcement that such bursts do occur. According to C. D. Howe, minister of trade and commerce, bursts of rods have occurred before at Chalk River and also in other reactors.

The released radioactive gases and dust from the burst rods were carried up the exhaust stack, but, owing to unusual atmospheric conditions and the severity of the discharge, they were not widely dispersed. Buildings near the reactor were contaminated sufficiently to affect research instruments. The premises were evacuated by all except those needed to take care of the damage, although the radiation level was not dangerous.

The reactor is now closed down indefinitely so that it can cool off and an examination be made. This has halted the production of isotopes in NRX. Other operations at Chalk River have been resumed.

According to Howe, such bursts must be expected when there is still so much to learn about the behavior of materials under high irradiation. One of the problems on which work is being done is the metallurgical changes produced by neutron bombardment.

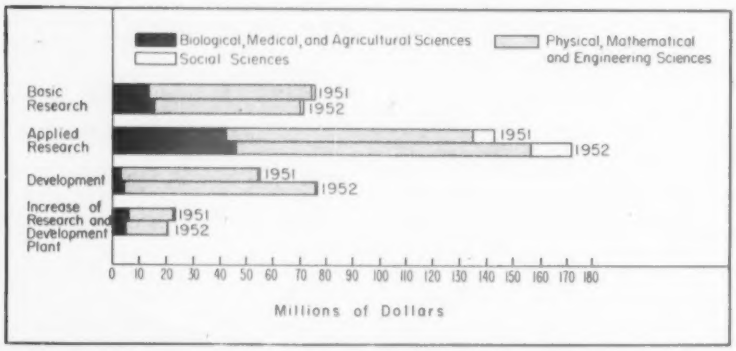
Declassify Power Reactor Data Says Gordon Dean

It is now time to give very serious consideration to the possibility of major declassification actions in the field of power reactors according to Atomic Energy Commission chairman Gordon Dean. This is necessary, Dean said in a speech at Brookhaven National Laboratory last month, if economical reactor power plants are to be built.

Dean gave as his reason for this point of view that "reactor technology has reached the stage of development where such extensive secrecy as we now have will almost certainly hamper progress, if it hasn't begun to already." The key to the problem of declassification, outside of purely nonmilitary fields, is, according to Dean, whether it will do more good than harm to place the information in the public domain.

Before now it did not seem that this question could be answered in the affirmative as far as the development of power producing reactors is concerned. Most of the technical problems that needed solving were being worked on behind the security barrier. Although problems might have been solved quicker without secrecy, Dean does not believe the improvement would have warranted declassification.

Now, however, Dean believes that the situation is changing. Technology has reached the stage, he feels, where some power reactors should be built, and they should be as economical as possible. According to Dean, this cutting of costs can best be done in a free competitive situation, and this requires a considerable lifting of the cloak of secrecy.


Biological, Medical, and Agricultural Sciences
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Social Sciences
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Increase of Research and Development Plant
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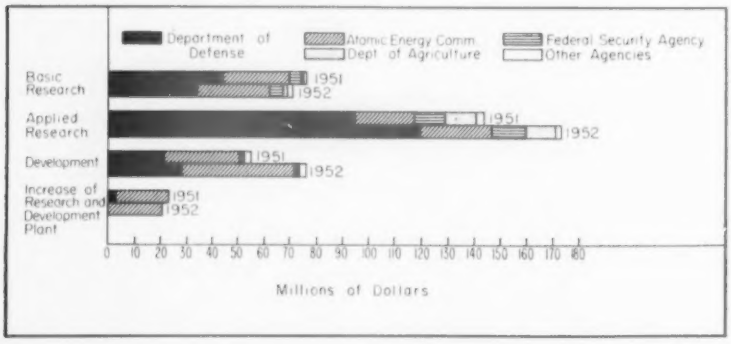
Millions of Dollars

FEDERAL FUNDS for scientific research and development at nonprofit institutions, by scientific fields



FEDERAL FUNDS for scientific research and development at nonprofit institutions, by agency and character of obligation

Research Support Studied by Science Foundation

According to a National Science Foundation study of federal support of research and development at nonprofit institutions, a total of $297-million was granted for fiscal 1951, and a total of $341-million was granted for fiscal 1952. Breakdown of the support is given in the two charts at left in terms of field of work and agency supporting the work.

Funds administered by the Department of Defense made up over 50% of the total in each year, compared with about 35% for the Atomic Energy Commission, 6% for the Federal Security Agency, and 5% for the Department of Agriculture. Fourteen other agencies accounted for less than 3% of the total.

Obligations for basic research totaled $76-million in 1951, and $71-million in 1952; for applied research, $144-million in 1951, and $173-million in 1952; for development, $54-million in 1951, and $77-million in 1952; and for increase in research and development plant, $23-million in 1951, and $20-million in 1952.

Agencies cooperating in the study in-

clude the Departments of Agriculture, Commerce, Defense, Interior, Labor, and State, the Atomic Energy Commission, Federal Civil Defense Administration, Federal Security Agency, Housing and Home Finance Agency, Library of Congress, National Advisory Committee for Aeronautics, National Security Resources Board, Office of Defense Mobilization, Reconstruction Finance Corporation, Tennessee Valley Authority, and Veterans Administration.

Atomic Project Raises $6-million for Research

The Michigan Memorial-Phoenix Project, conceived in 1948 as a memorial to University of Michigan war dead, has gone over the $6-million mark in its campaign for funds to support a program of research in peacetime applications of atomic energy. All of this money is from private contributions.

Research under the project got under way slowly in 1949 and 1950, and has continued to increase. Now, with the virtual completion of the campaign for funds, plans have been announced for the construction of a $1-million Phoenix Memorial Laboratory. A three-story 150 × 50-ft reinforced concrete building will be erected on the university's new North Campus development, a 267-acre site northeast of the main Ann Arbor campus.

The building will house such radiation sources as a betatron or van de Graaff, X-ray machines, gamma- and beta-ray sources, and possibly a neutron source. The university's cyclotron, and synchrotron, now installed on the main campus, may in the future be moved to the North Campus and be housed in a separate building.

Laboratory space will be available for work on the effects of high-energy particles on biological and physical systems, on tracer and radiation studies of solid state materials, on chemical procedures using large quantities of radioisotopes, and for new types of atomic energy research.

Basically, the work of the project is divided into five general categories. The first is the field of radiation-induced effects. The second is the field of tracers and tracer techniques.

The third category is the field of nuclear power. Activity in this field at present is mainly educational. In cooperation with the project, the university's engineering college and graduate school now offers an MS

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degree in nuclear engineering. The university has submitted a proposal and application to the Atomic Energy Commission for establishment of an educational and research reactor on the campus.

The last two categories are work in the social sciences. One of these is the general group of legal problems that have arisen during development of atomic energy under the McMahon act. The other is the group of social problems arising from civil defense needs, labor relations in the atomic energy program, and other related public administration problems.

To date, 54 individual projects have been organized, of which fourteen have been completed.

The planned project life is about ten years. In that time it is expected that sufficient new and important results originating from project research will justify the investment of more private funds and the continuation of the project indefinitely.

Curriculum Expanded for Third ORSORT Session

The curriculum for the third session of the Oak Ridge School of Reactor Technology, which is currently underway with a record enrollment of 81 students (NU, Nov. '52, p. 105), has been expanded over that for the first two sessions, and the faculty has been enlarged.

Seven subjects are taught in the course which is designed to provide a source of the uniquely trained technical personnel required for the Atomic Energy Commission's reactor development program. The subjects and instructors are:

Reactor Analysis and Mathematics, David K. Holmes and Lewis Nelson. The basic theory of reactors is developed in mathematical form in this course.

Reactor Chemistry and Technology, Sigfred Peterson and R. W. Stoughton. Chemistry and chemical technology of the separation processes related to operation of reactors are covered in this course.

Reactor Engineering, Lloyd G. Alexander, A. P. Fraas, and Paul C. Zmola. This course consists of a discussion of engineering principles of stress analysis, fluid dynamics, and heat transfer with application of these principles to reactors.

Reactor Materials, George E. Evans. This course covers the properties of many of the important materials used

in reactors together with the effects of radiation on these materials.

Reactor Nuclear Physics, Elda E. Anderson and Robert A. Charpie. This course provides the necessary background in nuclear physics for the understanding of the nuclear processes occurring in reactors.

Reactor Technology, Everitt P. Blizard, W. H. Jordan, and A. M. Weinberg. This course covers reactor theory and existing reactors, isotopes and some separation methods, use of radioisotopes in medicine, industry, and agriculture, and an over-all view of the activities of AEC. Also covered are shielding, principles and instrumentation of reactor controls, and the feasibility reports on some of the existing and proposed reactors.

Experimental Reactor Physics Laboratory, Herbert Pomerance and Paul H. Stelson. This course consists of about 25 experiments on fundamentals of nuclear particles and reactor measurements.

The fourth ORSORT session will begin next September. The deadline for applications is March 1. Application forms are available from ORSORT, Box P, Oak Ridge, Tenn.

IN BRIEF

• **3,400-curie source.** Ten cylindrical capsules containing a total of 2.5 lb of Co^{60} form the new gamma-ray generator at Knolls Atomic Power Laboratory, Schenectady. The source, which is kept at the bottom of a concrete pit and covered by 9 ft of water, will be used to study the effects of radiation on materials for the prototype nuclear power plant for submarine propulsion to be built at Knolls.

NUCLEAR NEWSMAKERS

Eger V. Murphree, member of the general advisory committee of AEC, will receive the 1953 Industrial Research Institute medal.

Gardner P. Wilson has been appointed senior development engineer at Consolidated Engineering Corp., Pasadena. Other new additions to the staff are **Wilson S. Brubaker,** senior research physicist, and **Paul Brock,** engineering mathematician.

John N. Butler has been appointed project engineer for the atomic energy research program at the University of Nevada. He was with the Bureau of Mines from 1945 to 1949.

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Output voltage, bias	0-150	.....	.....	0-150	0-150	.....
Output current, bias	0-5 Ma	.....	.....	0-5 Ma	0-5 Ma	.....
Ripple	10 mv	10 mv	10 mv	10 mv	10 mv	20 mv
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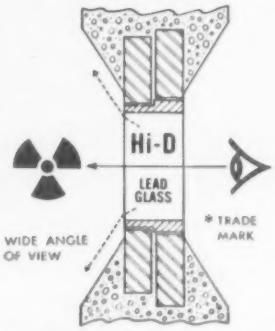

Hi-D Lead Glass Windows in concrete walls are dry, and hence are completely dependable. Where the viewing problem is mainly wide-angle vision, Hi-D Glass is fully competitive in price with zinc bromide.

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LETTERS

On Need for Radiation Society

DEAR SIR:

In a letter in the November '52 NUCLEONICS, p. 121, Mr. Saul Harris urged the formation of a radiation protection society.

I would like to point out that radiation protection is simply a new emphasized phase of the protection of industrial or occupational health. True, it has its own techniques for measuring hazard, but the medical and engineering phases of protection are not new. Neither is radiation hazard a new occupational exposure. It has been dealt with since application of the discoveries of Becquerel, the Curies, and Roentgen.

Much would be lost if those concerned with protection of occupational health were to split into many specialty groups. They can be most effective if they remain a closely knit team of cooperating specialists. They all have the same aim and the same general methods, differing only in technical details—they measure exposure, reduce exposure to tolerable levels, examine the man to verify success.

The American Industrial Hygiene Association is one forum where all the specialists can meet on a common ground. Neither the physician, the engineer, the chemist, the nurse, nor the physicist predominates. At least fifty health physicists are members. More would be welcomed.

We feel that it is appropriate for the Atomic Energy Commission to have its own health-physics association because of the classified information it deals with, but we see no need for those outside AEC to form their own association when they can have adequate voice in an existing association of people whose aim is the same as theirs and who are already using the tools they would specialize in.

HENRY F. SMYTH, JR.
Executive Secretary
American Industrial Hygiene Association
Pittsburgh, Pennsylvania

DEAR SIR:

Such an organization as Mr. Harris proposes was organized in Cincinnati area in April, 1952—in response to such needs as outlined in his communication.

The organization is known as the Cincinnati Radiation Society. Its

membership is composed of individuals who are interested in the use of any form of radiation, with the exception of strictly medical aspects. At present there are approximately 75 active members coming from a radius of about 50 miles. The objects and purposes of this organization are as follows:

1. To provide an organization for the discussion and dissemination of information concerning the theories, safe application, and implications of radiation and radioisotopes.
2. To encourage and assist those persons interested in ionizing radiation.
3. To improve standards relative to the use of ionizing radiation by exertion of our organized influence.
4. To promote education rather than legislative controls upon the use of radiation and radioisotopes.
5. To seek to declassify and/or make available information which might be of interest or direct use to members.
6. To provide a meeting place for those dedicated to the advancement of the peaceful applications of atomic and nuclear radiations.

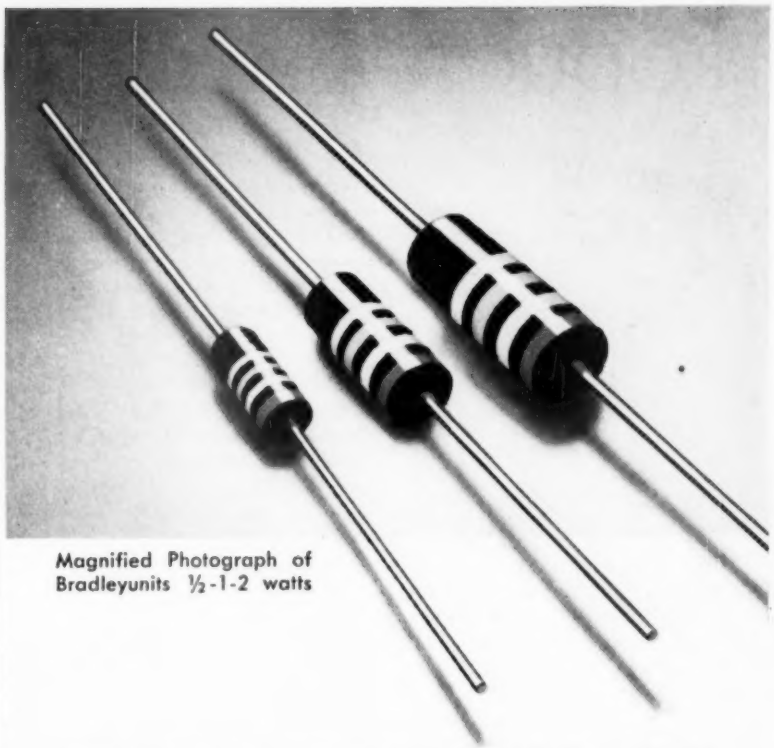
Meetings have been held at monthly intervals, and speakers have covered such topics as biological effects of radiation, use of radioisotopes in tool design, activation analysis, organization of health-physics programs, and planning of animal experiments utilizing radioactive isotopes. Seminars are being planned for instruction in various fields of radiation to be given to individuals who are entering this field.

There has been a great deal of interest shown in the Society, and the average attendance at the monthly meetings usually numbers about 50. On the basis of this local interest, we might assume that such an organization as proposed by Mr. Harris on a national level would serve a valuable purpose, especially in encouraging young men entering this rapidly expanding, multifaceted field.

EUGENE L. SAENGER, M.D.
Director, Radioisotope Laboratory
Cincinnati General Hospital
Cincinnati, Ohio

CORRECTION

In "Scintillation-Counter Safeguard for Co^{60} Source," by J. E. Dietrich and W. R. Kennedy (NU, Dec. '52, p. 74), the cable connections were numbered incorrectly. Terminals 2 and 3 of the photomultiplier probe should be interchanged.



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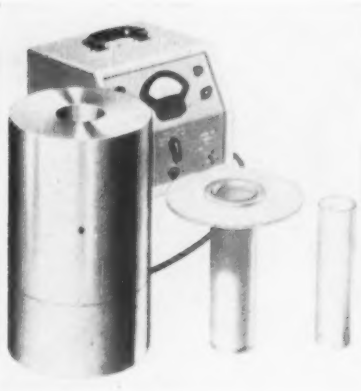


Gamma-Ray Detector

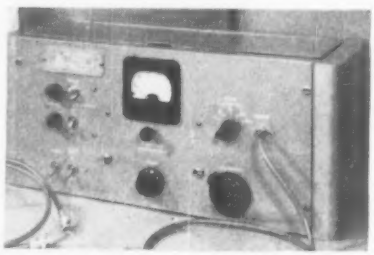
Welch Allyn, Inc., Skaneateles Falls, N. Y. The type GR gamma-radiation detector, referred to as the "Texas" G-M counter (NU, Oct. '52, p. 57), has a sensitivity of 40,000–50,000 cpm/μc for I^{131}. The well-type counter has an infinite life and is not harmed by overvoltages. It can be connected directly to standard recorders. (P-1)

Scaler-Printer

North American Philips Co., Inc., Research and Control Instruments Division, 750 S. Fulton Ave., Mount Vernon, N. Y. This binary scaler-printer records in printed numerals up to 999. Six binaries may be cascaded to provide scale-of-2 to scale-of-64 operation. An interval timer allows count accumulation for a predetermined interval up to 55 sec, reproducible to within 0.13 sec. With a negative input pulse of 1-volt amplitude, the maximum counting rate is about 200,000 pulses/sec. The unit is available with a capacity of 2 channels of 64,000 counts each; it can be modified for channel analyzing with selection of cascading of up to 6 channels with a capacity of 10 counts each. (P-2)

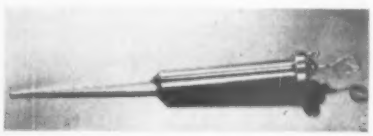
Labeled Chemicals

Radioactive Products, Inc., 443 W. Congress, Detroit 26, Mich. Ring-labeled cholestenone-4-C^{14} is available with a specific activity of 10 μc/mg. Minimum quantity is 10 μc. Also available is progesterone-4-C^{14} with a specific of 10 μc/mg. (P-3)



Pulse Generator

Radiation Counter Laboratories, Inc., 5122 W. Grove St., Skokie, Ill. The mark 15 model 47 pulse generator has a pulse rise time of less than 10^{-8} sec and fall time of 350 μsec. Ranges are 1, 3, 10, 30, and 100 mv pulse amplitude, with 10-turn linear potentiometer control over these ranges. Pulse height is standardized against a cell in the instrument. Repetition rate is 3,600 pulses/sec. (P-4)



G-M Probe Counter

H. W. Leighton Laboratories, 26 Herman St., Glen Ridge, N. J. The G-M probe-counter needle is 10 cm long with a diameter of 2 mm; the sensitive volume starts 5 mm from the tip and extends 18 mm back from the tip. Starting voltage is 1,050 volts $\pm 10\%$. Plateau length is 75–100 volts; the slope is $< 10\%$ per 100 volts. (P-5)

Photomultiplier Shield

James Millen Mfg. Co., Inc., 150 Exchange St., Malden, Mass. The 80802B magnetic shield is designed for shielding equipment that uses the RCA 5819 photomultiplier tube. The material is hydrogen-annealed 0.050 mu-metal. Centering ring minimizes leakage between tube and shield. (P-6)

Decontaminating Agent

Enley Products, Inc., 254 Pearl St., New York, N. Y. This surface-decontaminating agent, called CP Cleaner, holds in suspension ions, dirts, and greases. It is claimed to dissolve instantly in soft, hard, and salt water. Almost neutral *p*H is said to allow its use on the human body. (P-7)

Universal Balance

Voland and Sons, Inc., Box 680, New Rochelle, N. Y. The model 860D universal balance may be used as a

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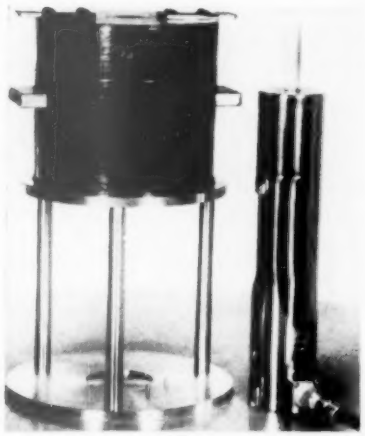
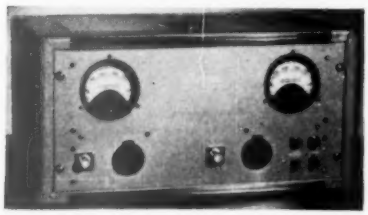
high-speed projection balance or, by means of a sensitivity shift, as an analytical balance. As a projection balance, readings to 0.1 mg for 200-gram loads are claimed to be obtained in seconds; as an analytical balance, sensitivity is 0.01 mg for 100-gram load. All weights are obtained by selector knobs. (P-8)



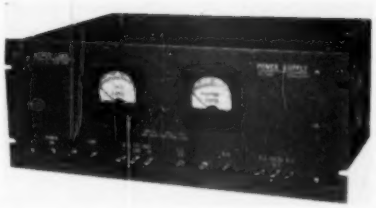
Scintillator Counter

N. Wood Counter Laboratory, 5491 Blackstone Ave., Chicago 15, Ill. The type SC-2L scintillation counter has a well-type Na(Tl) crystal into which sample vials may be inserted. The counter has a 300-volt plateau with less than 3% rise in counting rate. The lead shield is 2 in. thick and extends 2 in. above and below the 2-in. crystal. (P-9)



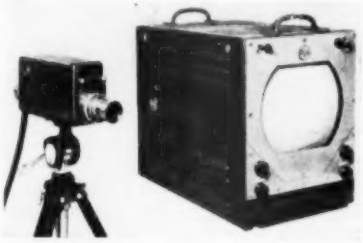
Dual High-Voltage Supply

Radiation Instrument Development Laboratory, 2337 W. 67th St., Chicago 36, Ill. The model 80 dual high-voltage supply, consisting of two individually regulated sources, is designed for applications such as coincidence counting. Available ranges are 500–1,000 volts negative, 500–2,500 volts positive, and 600–5,000 volts positive. Regulation is 0.01% per 1% change in line voltage. Any combination of these ranges can be ordered. (P-10)



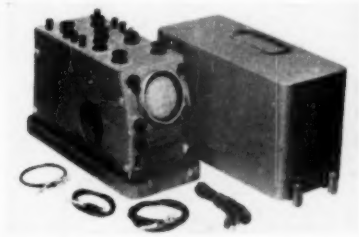
Regulated Power Supply

Kepco Laboratories, Inc., 131-38 Sanford Ave., Flushing 55, N. Y. The model 141 power supply provides 100–400 volts, d-c, continuously variable at 150 ma max. Output voltage variation is less than ½%, ripple is less than 5 mv, and impedance is less than 2 ohms. Filament supply is 6.3 volts, a-c, at 10 amperes. (P-11)



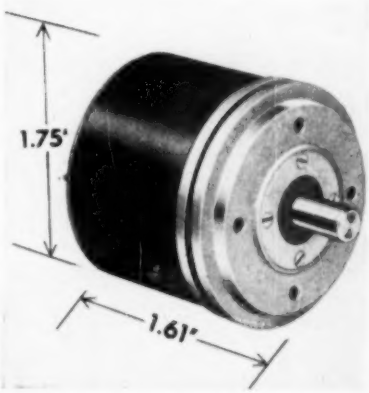
Television System

RCA Victor Division, Radio Corp. of America, Camden, N. J. The model 1TV-5 closed-circuit television system consists of a 7-lb camera and a 70-lb combination monitor-power supply-control unit. The image is conveyed from camera to screen by a camera cable; equipment can also be operated with microwave transmission over extended distances. Since scanning frequency is the same as standard broadcast TV, home television sets can be used as auxiliary viewers. (P-12)



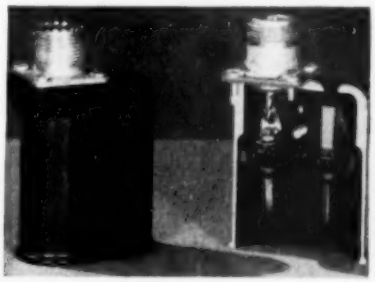
Portable Oscilloscope

Hickok Electrical Instrument Co., 1058 Dupont Ave., Cleveland 8, Ohio. Using a type 3RP-1 cathode-ray tube and d-c amplifiers, the model 380 portable oscilloscope provides vertical amplification to 2.5 Mc, and horizontal amplification to 100,000 cycles. The sweep-circuit oscillator has a range of 3–50,000 cycles. Horizontal and vertical deflection sensitivity is 0.1 RMS volts/in.; vertical and horizontal direct-input sensitivities are 22 and 28 RMS volts/in., respectively. The instrument weighs 14 lb. (P-13)



High-Resolution Potentiometer

G. M. Giannini and Co., Inc., East Orange, N. J. The Spiralpot slidewire potentiometer is available in 500-ohm ranges from 500 to 2,500 ohms, with unlimited (zero increment) resolution. It has high resolution with angular shaft displacements up to 3,600 degrees. It performs in the range −55° to +71° C and dissipates up to 2 watts at +71° C. Less than 2 oz-in. of torque is required. The Spiralpot is claimed to last over a million revolutions at 200 rpm. (P-14)



Terminal Adapter

Allen B. Du Mont Laboratories, Inc., Instrument Division, 1500 Main Ave., Clifton, N. J. The type 2592 terminal adapter is designed for instruments, such as cathode-ray oscillographs, having banana-jack-type terminals spaced ¾ in. on centers. Built into the adapters are impedance-matching re-

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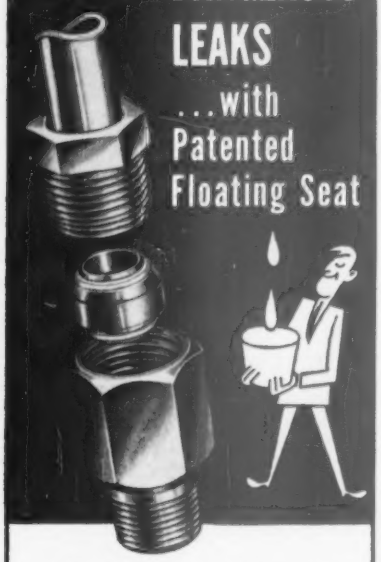

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sistors. Available are 52, 75, and 93-ohm adapters for use with UHF and BNC coaxial connectors, and 52 and 75-ohm adapters for use with type N coaxial connectors. (P-15)

D-C Power Supply

Neutronic Associates, 83-56 Vietor Ave., Elmhurst 73, N. Y. The model 33HRR regulated, d-c supply provides 1–30 kv at 5 ma. Regulation is 0.1% at all voltages; output is reversible in ranges of 0–5, 0–15, and 0–30 kv. Line-voltage stabilization is 0.1% from 105 to 130 volts; ripple is said to be less than 0.05% of d-c output. (P-16)

Torque Balances

Ohaus Scale Corp., 1050 Commerce Ave., Union, N. J. These torque balances are available with capacities from 1 mg to 20 gm, with sensitivities of 0.002 to 0.02 mg. Rapidity of weighing is increased by a special

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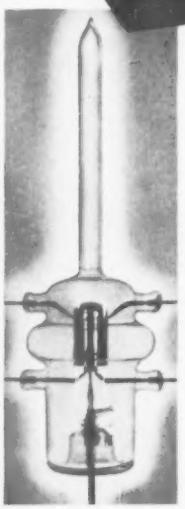

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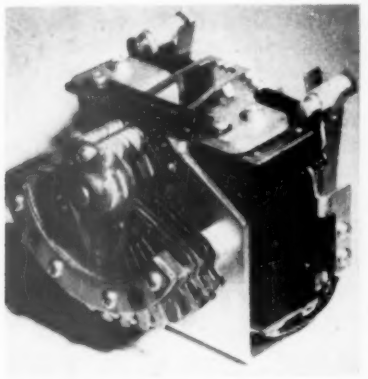
damping device; stand permits specific-gravity weighing. The balances have eye-level reading and are resistant to shock. (P-17)

Variable Delay Line

Advance Electronics Co., Box 394, Passaic, N. J. The type 507 miniature delay line is continuously variable from 0 to 0.8 μsec. Rise time is $0.0012\sqrt{t}$ μsec, where t is the delay in millimicroseconds; characteristic impedance is 390 ohms, nominal. Attenuation, in db per 100-mμsec delay, is: zero below 3 Mc, 0.5 at 8 Mc, and 1 at 15 Mc. Its size is $1 \times 4 \times 4$ in. (P-18)



Stepping Switch

Consolidated Engineering Corp., 300 N. Sierra Madre Villa, Pasadena 8, Calif. The model 27000 self-interrupted, bidirectional stepping switch is applicable to high-speed differential counters and servomechanisms. It is convertible to a stepping motor for special servo and control applications. External pulses of short duration can be applied to advance or reverse the shaft and wipers one step for each pulse. Banks of up to 25 contact positions can be furnished, or full 360-

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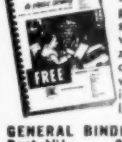

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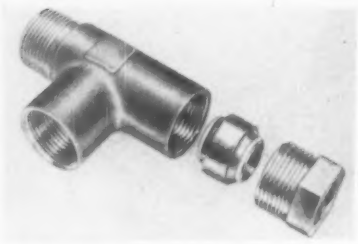
degree rotation can be obtained over 36 contacts. (P-19)



Stainless-Steel Fittings

Special Screw Products Co., 5445 Dunham Rd., Bedford, Ohio. These stainless-steel flared-tube fittings permit different sizes of tubing or tube-pipe combinations to be joined in a single fitting. These couplings, available in any combination up to 1 in., eliminate the need for supplementary reducing couplings. Connectors, unions, tees, and elbows incorporate an automatically self-centering seat which provides a wedge-type seal, even though assembled off center. Teflon rings double-seal the seat. (P-20)

INDUSTRY NOTES

▶**Tracerlab, Inc.,** has elected William A. Kerr general manager. His headquarters are at 2295 San Pablo Ave., Berkeley, Calif.

▶**Nuclear Research and Development, Inc.,** 6425 Etzel Ave., St. Louis 14, Mo., has appointed Marshall R. Cleland as a senior research physicist. He will head several industrial research projects.

▶**Nuclear Instrument and Chemical Corp.,** 229 W. Erie St., Chicago, Ill., has elected Dr. Harry K. Ihrig, John Cleaver, and James M. Phelan to the board of directors.

▶**High Vacuum Equipment Corp.,** 349 Lincoln St., Hingham, Mass., has been organized for development, design, and manufacture of high-vacuum equipment.

▶**Wakefield Industries, Inc.,** 5108 W. Grove St., Skokie, Ill., has announced the acquisition of Del-Mar Scientific Glass Co. All Del-Mar personnel have been transferred to Wakefield's new plant in Skokie.

▶**Beckman Instruments, Inc.,** South Pasadena, Calif., has appointed John F. Bishop assistant general manager of the instrument division.

LITERATURE AVAILABLE

Servo components. Brochure lists instruments, including amplifiers, analyzer, null detector, millivoltmeter, and motor control. *Industrial Control Co., Wyandanch, L. I., N. Y.* (L-1)

Vacuum-tube electrometers. Bulletin describes electrometers and accessories. *Keithley Instruments, 3868 Carnegie Ave., Cleveland 15, Ohio.* (L-2)

Power supplies. Brochure gives line of low and high-voltage power supplies. *Kepco Laboratories, Inc., 131-38 Sanford Ave., Flushing 55, N. Y.* (L-3)

Electronic equipment. Pamphlet gives data on laboratory electronic equipment. *Daven Co., Dept. LE, 191 Central Ave., Newark 4, N. J.* (L-4)

Subminiature tubes. Booklet describes 11 subminiature cathode-type tubes, including 6 new tubes. *Raytheon Mfg. Co., Technical Information Service, Special Tube Section, 55 Chapel St., Newton 58, Mass.* (L-5)

Laboratory furniture. Catalog 25 describes unitized steel furniture. *Fisher Scientific Co., 711-723 Forbes St., Pittsburgh 19, Pa.* (L-6)

Radiation equipment. Catalog W-52 describes X-ray accessories and radiation-protection items. *Bar-Ray Products, Inc., 209 25th St., Brooklyn 32, N. Y.* (L-7)

Cobalt-60 needles. Booklet no. 46 describes Co^{60} interstitial needles and accessories. *Tracerlab, Inc., 130 High St., Boston, Mass.* (L-8)

Laboratory equipment. "What's New for the Laboratory" lists 27 new items for laboratory research. *Scientific Glass Apparatus Co., Inc., Bloomfield, N. J.* (L-9)

Industrial freezers and testing units. Folder gives data on freezers and complete-temperature-range testing units. *Industrial Freezer Division, Webber Appliance Co., Inc., 2740 Madison Ave., Indianapolis 3, Indiana.* (L-10)

Reactor-control equipment. Price list no. 12 describes control equipment for nuclear reactors. *Radiation Counter Laboratories, Inc., 5122 W. Grove St., Skokie, Ill.* (L-11)

Laboratory aids. Catalog no. 2 includes laboratory lubricants, cements, paints. *Varniton Co., 416 N. Varney St., Burbank, Calif.* (L-12)

Every Top Management Man... In Every Industry

SHOULD BE ABLE TO ANSWER THESE QUESTIONS ABOUT A MOST CRITICAL EMERGENCY IN OUR COUNTRY'S AFFAIRS

Q. Why is iron and steel *scrap* a matter of importance to me?

A. Steel for our country's military program and civilian economy is being produced at the annual rate of 107,000,000 tons in 1951 . . . 119,500,000 tons expected in 1952. Steel-making capacity is being increased now to meet those quotas.

What Do I Get For My Scrap?

In addition to being paid for your scrap, you remove nuisance inventory from your plant—saving valuable floor space. Also, you have a better chance of *getting* new steel or steel products. But, *most important*—you help alleviate a dangerous condition threatening our country's capacity to rearm and satisfy civilian requirements at the same time.

Q. How does scrap figure in the production of steel?

A. Steel is composed, generally speaking, 50% of pig iron, 25% of "production" scrap (that is, the scrap which is produced as a by-product of steel-making) and 25% of "purchased" scrap.

Q. Is scrap getting scarce?

A. Yes. The supply of *purchased* scrap is not increasing fast enough to meet the needs of increasing steel production.

Q. What if the needed scrap isn't obtained?

A. Open-hearth furnaces will not be able to operate at capacity. That will mean a loss of steel production . . . and fewer products made of steel.

Q. Why not use pig iron instead of scrap?

A. Every ton of scrap conserves approximately 2 tons of iron ore, 1 ton of coal, nearly ½ ton of limestone and many other vital natural resources—to say nothing of the extra transportation facilities that would be otherwise required.

Q. How can *more* scrap be furnished?

A. By everybody pitching in—as we always do in every emergency—and searching out all possible sources of scrap.

Q. What are these sources?

A. Metal-fabricating plants normally turn over to scrap dealers the scrap left from machining. But there's not enough of this to fill our present enormous need. So everybody—both in and out of the metal-fabricating industries—must sell scrap in the form of *idle metal.*

Every pound of idle metal is needed to keep our steel mills operating at top capacity. Sell your idle metal to a local scrap dealer right away.

What Do I Do *First*?

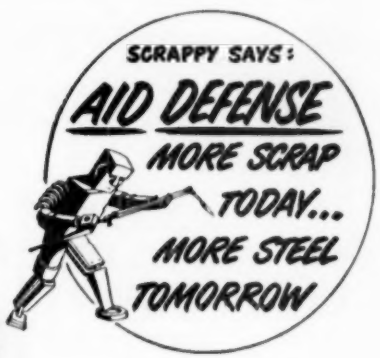
Write for free booklet. It tells how to set up a Scrap Salvage Program in your plant. Thousands of plants are cooperating. Do *your* part now! Address Advertising Council, 25 West 45th Street, New York 19, N. Y.

Q. *We* don't *produce* scrap—how can *we* help?

A. Scrap is any kind of iron and steel that's gathering dust—obsolete machines or structures, jigs and fixtures, pulleys and wheels, chains and track, valves and pipe—*anything* with rust on it or dust on it. Non-ferrous scrap is needed, too.

Q. What do we do with it when we find it?

A. Use your normal channels or get in touch with a recognized scrap dealer.



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The finest dielectrics, especially for high frequency, high temperature service. Won't carbonize under arcing. Won't DC plate. Zero water absorption by ASTM test. Unaffected by extreme humidity. Chemically inert, non-gassing, immune to corrosive atmospheres, fungus, oil, solvents. Non-flammable, tough, resilient, withstand and absorb mechanical shock and vibration.

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H. E. Hilty, Mgr.

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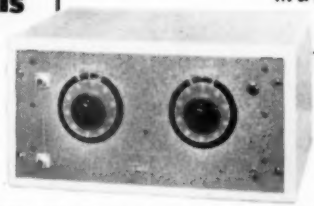
This index is published as a convenience to the readers. Every care is taken to make it accurate, but NUCLEONICS assumes no responsibility for errors or omissions.

For Precise pulse height analysis

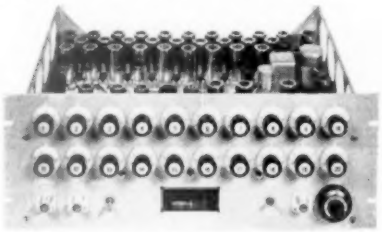
ATOMIC'S Single and Multiple Channel Analyzers

MODEL 510 SINGLE CHANNEL DIFFERENTIAL PULSE HEIGHT ANALYZER

for pulse amplitude distribution studies in the range from zero to 100 volts. When used with a Model 204 Linear Amplifier and any of Atomic's Scalers, this instrument will enable laboratories to study the energy distribution of nuclear particles.



The channel width or "window" is settable between zero and seven volts. Any input pulse, whose maximum amplitude falls between these channel limits, will be transmitted to the output.



Model 520 Pulse Sorter Component

MODEL 520 MULTIPLE CHANNEL DIFFERENTIAL PULSE HEIGHT ANALYZER

enables the nuclear physicist to take accurate data on the amplitude distribution of electrical signals, having random amplitude and random time distributions. The Model 520 speeds the taking of spectrum data on long life isotopes and provides data on short lived isotopes with fewer decay corrections.

Each of the 20 channels has a scale of sixteen and register, plus the total count and surplus count scalers; maximum counting rate per channel — 320 C.P.S. (regularly spaced). Resolving time — both of two pulses 2.3 microseconds apart, or closer, are rejected. Channel resolution time — 15 microseconds.

The model 520 can readily be adapted to the gathering of data on the distribution of fog particle sizes, distribution of radar receiver signal amplitudes and other similar random distribution studies.

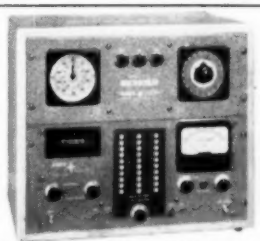
LINEAR AMPLIFIERS: Models 204B and 204C

For use with proportional counters, ionization chambers and in scintillation counting to amplify small pulses up to a level which can be observed on an oscilloscope and counted by means of standard scalers or counting rate meters. Models 204B and 204C have the following specifications in common: *Gain control* is achieved by coarse attenuator having 6db steps to 30 db. *Input circuit* is designed to be driven by low impedance source, or amplifier can be used alone, accepting either positive or negative input signals between 200 microvolts and 1 millivolt. *Two outputs*, low level drives coaxial cable to maximum 5 volts at 150 ohms; high level provides maximum signal of about 100 volts without appreciable amplitude distortion. *Precision calibrated pulse amplitude discriminator* operates at high level output, and all pulses above set level cause discriminator to give out 10 volt 0.4 microsecond pulses.

Model 204B has a three-position "rise time" switch for selectable pulse response; also includes fine gain control.

Model 204C has pulse response determined by input time constant settings from 0.6 to 16 microseconds in seven steps, and rise time settings from 0.2 to 2 microseconds in five steps.

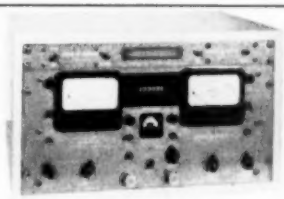
SCALERS, Manual or Automatic, Binary or Decade



Model 1070A (shown) — latest version of the first complete automatic scaler — scale of 1000 or 4096 — choice of 5 microsec. or 1 microsec. resolution — preset time — preset count — scale selector — electric register — pulse amplifier — discriminator — regulated 400 to 2500 volt supply. Provision for remote automatic operation and many other features. Atomic offers a choice of complete binary or decade scalers to meet every laboratory requirement.

Also, separate regulated High Voltage Supplies, either polarity.

ANTICOINCIDENCE ANALYZERS



Model 503C Two Channel Anticoincidence Analyzer — a complete GM tube anticoincidence analyzer with two separately adjustable line and load regulated H.V. Supplies and drag voltage for using the Libby Kulp Screen-Wall Counter. When used with multi-tube "reject" input and special shielding the 503C permits observations of extremely low counting rates with high backgrounds. Model 503C has a built-in scale of 16 and a totalizing register with register drive circuit included.

Model 503B — similar to 503C but without scale of 16 — for use with external scaler or at very low counting rates.

For complete details on these and other of Atomic's instruments, write today for illustrated, indexed catalog.

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PORTABLE — LOW COST — VERSATILE

for LABORATORY Counting and Standby

The low cost of **nuclear's** Model 180 Rediscaler makes it a logical choice for an "extra" unit, yet its construction to regular **nuclear** standards qualifies it for precise laboratory work. There is no "skimping" on quality and all essential scaler features are included. Model 180 has ¼-volt G-M input sensitivity—decade indication—timer outlet—adjustable high voltage well stabilized—less than .05% change at 900 volts for line voltage change of 1% between 95 and 130 volts.

for MEDICAL Diagnosis and Research

Small size (only 10" x 9½" x 13¾") of Model 180 makes it easily portable—thereby available for many uses by medical technicians and research workers. Usable with standard G-M sample counters, probes, and liquid sample counters for clinical applications such as tumor location, blood volume determination, and urine analysis.

for INSTRUCTION and Training

Demonstrating radioactivity counting with Model 180 Rediscaler keeps explanations clear and simple—gives instructors an excellent tool for basic education as well as a practical laboratory instrument. Whether just beginning, or expanding radiolaboratory facilities, the Model 180 is an excellent unit for new radiotechnicians. You can depend on the Model 180 Rediscaler—it's covered by **nuclear's** exclusive one-year warranty.

NEW nuclear 1953 Catalog!

Make the 1953 **nuclear** catalog your all-need reference source for modern **nuclear** instruments, radiochemicals, and accessories. New scalers, counters and detectors all are completely described and illustrated . . . Your copy sent promptly on request—write for it today.

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